Nature and Extent of Releases at the INTEC Fuel Reprocessing Complex

April 2006

Idaho Cleanup Project

Nature and Extent of Releases at the INTEC Fuel Reprocessing Complex

April 2006

Idaho Cleanup Project

Idaho Falls, Idaho 83415

ABSTRACT

This report documents past releases that have occurred within and under the Idaho Nuclear Technology and Engineering Center Fuel Reprocessing Complex. During fuel reprocessing operations, some leakage of process liquids occurred, allowing contamination to be released to the subsurface. The contaminated soils underlying the Fuel Reprocessing Complex fall within the scope of Operable Unit 3-13 Group 2 (soils under buildings) and are covered under the *Final Record of Decision Idaho Nuclear Technology and Engineering Center Operable Unit 3-13*, which states that the selected alternative remedy for Group 2 was deferred action, pending building deactivation, decontamination, and decommissioning activities. Information about the release source terms is necessary to support the goal to initiate deactivation, decontamination, and decommissioning of this building using the Comprehensive Environmental Response, Compensation, and Liability Act process. This report provides information on the nature and extent of contamination sources for use in the development of the non-time-critical removal action and engineering evaluation/cost analysis risk assessments.

CONTENTS

FIGURES

TABLES

ACRONYMS

bgl   below ground level

CERCLA   Comprehensive Environmental Response, Compensation, and Liability Act

CPP   Chemical Processing Plant

DD&D   deactivation, decontamination, and decommissioning

DOE   U.S. Department of Energy

DOG   dissolver off-gas

FAST   Fluorinel Dissolution Process and Fuel Storage

FRC   Fuel Reprocessing Complex

HCL   Hot Chemistry Laboratory

HPP   Headend Process Plant

HWMA   Hazardous Waste Management Act

ICDF   Idaho CERCLA Disposal Facility

INEEL   Idaho National Engineering and Environmental Laboratory

INL   Idaho National Laboratory

INTEC   Idaho Nuclear Technology and Engineering Center

MCC   Multi-Curie Cell

NWCF   New Waste Calcining Facility

OU   operable unit

PEW   process equipment waste

PEWE   process equipment waste evaporator

PM   process makeup

RAF   Remote Analytical Facility

RCRA   Resource Conservation and Recovery Act

Rover   Space Nuclear Propulsion Program

TBP   tributylphosphate

TFF   Tank Farm Facility

USC   *United States Code*

VOG   vessel off-gas

Nature and Extent of Releases at the INTEC Fuel Reprocessing Complex

1. INTRODUCTION

This report documents past releases that have occurred within and under the Fuel Reprocessing Complex (FRC) and the CPP-602 laboratory located at the Idaho Nuclear Technology and Engineering Center (INTEC) on the Idaho National Laboratory (INL) Site (Figure 1-1). The FRC is composed of three adjoining buildings that were used for reprocessing spent nuclear fuel to recover uranium: (1) Fuel Reprocessing Building (CPP-601), (2) Remote Analytical Facility (RAF; CPP-627), and (3) Headend Process Plant (HPP; CPP-640). The CPP-602 laboratory was used to support operations at the FRC and is located north of and adjacent to CPP-601, sharing a common wall and utilities. For the remainder of this report, the CPP-602 laboratory will be considered part of the FRC. During fuel reprocessing operations, some leakage of process liquids occurred, allowing contamination to be released to the subsurface. This report provides background information on the system associated with the release, a description of the release, and previous investigations and cleanup activities performed. Estimates of waste volume released, including curie contents associated with the release, also are provided.

The contaminated soils underlying the FRC fall within the scope of Operable Unit (OU) 3-13 Group 2 (Soils Under Buildings) and are covered under the *Final Record of Decision Idaho Nuclear Technology and Engineering Center Operable Unit 3-13* (DOE-ID 1999), which states that the selected alternative remedy for Group 2 was deferred action, pending building deactivation, decontamination, and decommissioning (DD&D) activities. The post-DD&D remediation goals are to prevent exposure to future workers and residents and to minimize possible leaching and transport of contaminants to the underlying Snake River Plain Aquifer. The remediation goals will be met by implementing institutional controls and either (1) capping the contaminated areas to restrict access and prevent unacceptable groundwater risk or (2) excavating contaminated soils that exceed the remediation goals and subsequent disposal at the Idaho CERCLA Disposal Facility (ICDF).

Information about the contaminant-release source terms is necessary to support DD&D of the FRC under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) process. This report provides the information on the nature and extent of contamination sources for use in the development of the non-time-critical removal action and engineering evaluation/cost analysis risk assessments.

This report presents source terms descriptions that include the possible type of waste, an estimate of the volume released to the environment, and an estimate of the concentration of certain radionuclides (Cs-137, Sr-90, I-129, H-3, and Tc-99). These radionuclides are contaminants of concern for the INTEC tank farm, which was the destination for most of the FRC waste streams. For the INTEC tank farm, other contaminants such as inorganics (metals) and organics were found to be of little or no risk (DOE-ID 1999) compared to the risk associated with the previously identified radionuclides. The estimated volume of the releases and associated source terms are in large part based on knowledge of the waste stream, not on analytical data that are specific to each release. The estimates presented are considered conservative estimates.
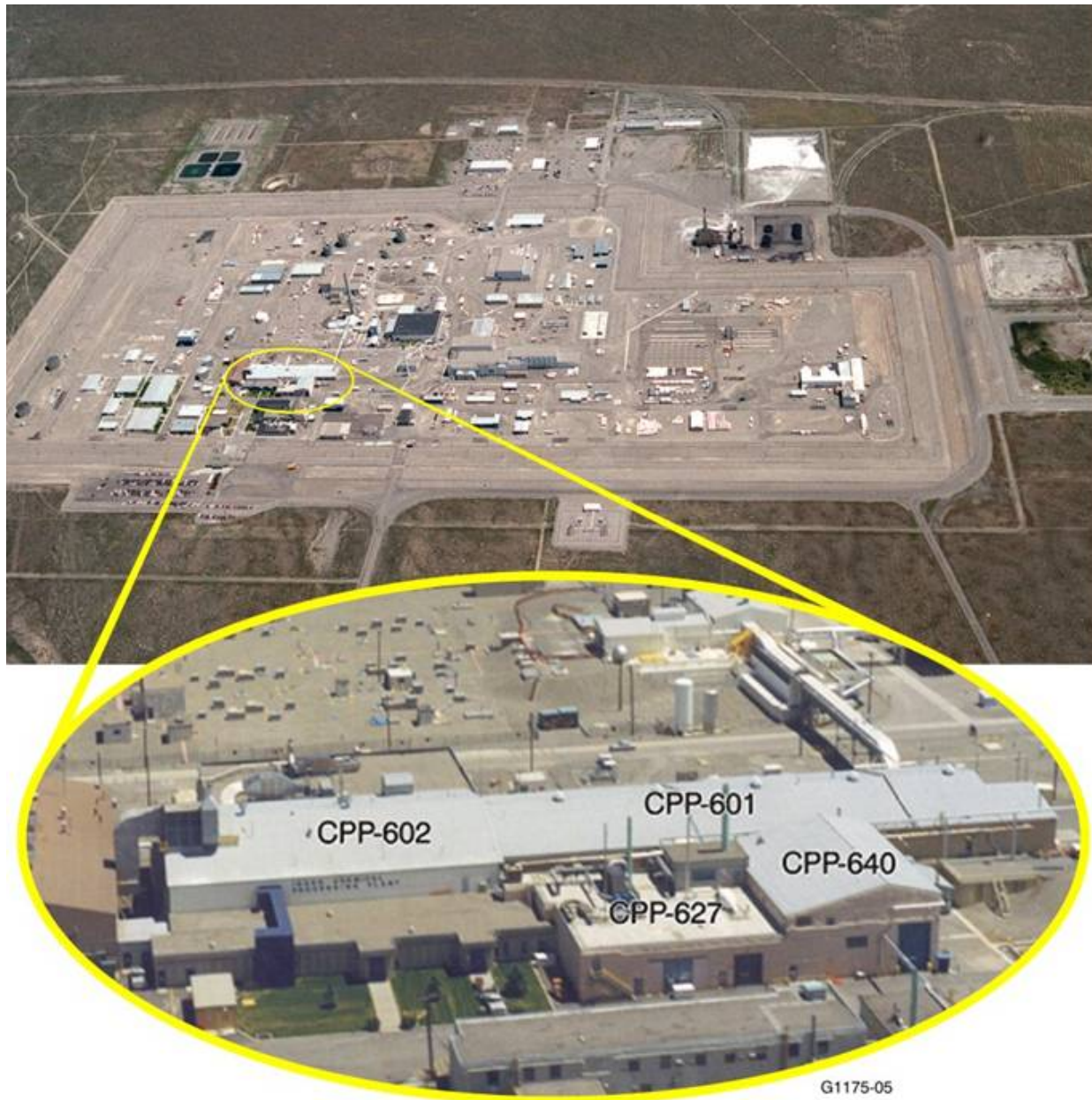
Figure 1-1. Location of the Fuel Reprocessing Complex within the Idaho Nuclear Technology and Engineering Center.

To understand better the releases that occurred, a good understanding of the FRC and its systems is required. Section 2 provides an overview of the structure of the FRC. Section 3 provides the necessary information to understand how the fuel was reprocessed and information on the systems and infrastructure that supported the reprocessing campaigns. Section 4 includes the detailed descriptions of the releases and the estimated source terms. Section 5 summarizes the releases to provide a total source term for the FRC.

2. STRUCTURE OF THE FUEL REPROCESSING COMPLEX

2.1 Building Configurations

This section provides a general description of the construction and layout of the various FRC buildings.

2.1.1 CPP-601 Fuel Reprocessing Building

The CPP-601 facility is essentially rectangular (244 × 102 ft) and consists of six levels with the primary portion located below ground. The top level is above grade and contains an open area that was used for transfer of fuel elements to the process equipment and for chemical storage, makeup, and transfer. As illustrated in Figure 2-1, CPP-601 is divided into the following six floor levels, beginning with the top level and descending to the bottom level:

- Process makeup (PM) area floor at elevation 4,927.5 ft (+10.5 ft above grade)

- Operating floor (PO) at elevation 4,917.0 ft (grade level)

- Service area floor (PT) at elevation 4,905.3 ft (11.7 ft below grade), which includes the vent tunnels (VT, SJ, SK)

- Access corridor floor (PA) and cell floor at elevation 4,895.0 ft (22 ft below grade)

- Control room (WA & WF) and cell floor at elevation 4,885.0 ft (32 ft below grade), which includes the waste trench and pump pits (WB, WC, WD, WE)

- Tank vaults (WG & WH) at elevation 4,859.5 ft (57.5 ft below grade).

The WG/WH tanks (known as the CPP-601 deep tanks) collected waste generated during fuel reprocessing from the CPP-601 separation processes. Other sources could send waste to these tanks, including the chemistry labs in CPP-602, CPP-627, and CPP-684. These sources entered the system through lab drains. Process systems in CPP-627 and CPP-666 also could send waste to the WG/WH tanks. CPP-640 had its own waste collection tanks (three 500-gal tanks designated VES-HW-100, VES-HW-101, and VES-HW-102) located in tank vaults at the lowest level of the CPP-640 building. These tanks sent waste to tanks located in CPP-641 until 1991 when a line was installed allowing transfer to CPP-601.

2.1.2 CPP-602 Laboratory Building

The CPP-602 laboratory facilities are primarily analytical and developmental facilities for chemical and radiochemical analyses for bench-scale development work. The analytical and developmental facilities include conventional chemical laboratories as well as radiochemistry laboratories for small quantities of radioactive materials. The denitrator process, which converted liquid uranium product from CPP-601 to a dry solid for shipment, was located in the CPP-602 basement.

1. Elevator
2. Cell hatch (typical)
3. Access chute (typical)
4. Truck stop (typical)
5. East H&V room
6. Loading ramp

7. Ramp (vehicle)
8. Foul weather entry
9. West H&V room
10. SOG filter room
11. SOG blower room

12. West sample corridor
13. East sample corridor
14. Operating corridor
15. North corridor
16. South corridor
17. Service corridor

18. West vent corridor
19. East vent corridor
20. South vent corridor
21. High vacuum cell
22. Ramp (personnel)
23. Fill

24. Access corridor
25. South access corridor
26. Waste trench
27. West control room, WA
28. East control room, WF
29. Pump cell, WB

30. Pump cell, WC
31. Pump cell, WE
32. Pump cell, WD
33. West tank cell, WG
34. East tank cell, WH

PM area
El. +10'-6"

Operating area
El. 0'-0"

Service area
El. -11'-8"

Access corridor
and cells
El. -22'-0"

Control rooms
and cells
El. -32'-0"

Tank vaults
El. -57'-6"

G1175-01

Figure 2-1. CPP-601 expanded view showing building levels and internal corridors, cells, and rooms.

2-2

The CPP-602 laboratory has a rectangular floor plan approximately 150×144 ft and extends from approximately 14.5 ft below grade to 14.5 ft above grade at the second-floor level. A cement pipe trench for the liquid waste headers lies under the basement floor to carry waste from the laboratories. A pair of stainless-steel pipe headers transports process equipment waste (PEW) containing chemical and radioactive waste to the CPP-601 deep tanks for storage prior to processing through the INTEC process equipment waste evaporators (PEWEs).

2.1.3   CPP-627 Remote Analytical Facility

CPP-627 is a 13,727-ft^2, two-story facility entirely above ground and is adjacent to and attached to CPP-601. This facility was constructed in 1955 to house analytical, experimental, and decontamination facilities. The building was constructed using reinforced concrete and masonry block, and it is currently inactive and undergoing demolition. CPP-627 included the Hot Chemistry Laboratory (HCL), the Old Shift Laboratory, the RAF, the Multi-Curie Cell (MCC), the Emission Spectroscopy Laboratory, and the Decontamination Development Laboratory. The custom dissolution process was located in the HCL and MCC.

The northern third of the building housed the analytical facilities. The RAF occupied the ground floor and consisted of two lines of shielded gloveboxes for remote sample preparation and analysis. The Old Shift Laboratory occupied the second floor and provided bench and hood space for chemical analyses. Analytical services were provided around the clock to support fuel-processing operations.

The middle third of the building was originally a high bay decontamination facility, providing space for water and chemical cleaning of contaminated equipment. The decontamination facility was replaced in 1980 by the decontamination cell at the New Waste Calcining Facility (NWCF). The space was converted to house the Emission Spectroscopy Laboratory and Decontamination Development Laboratory. The second story provided a fan and filter loft for handling off-gas from radioactively contaminated portions of CPP-627.

The southern third of CPP-627 contained experimental facilities, the HCL and the MCC. The HCL consisted of lab benches, hoods, and shielded gloveboxes. The MCC was designed for experiments using fully irradiated fuel. Both the HCL and the MCC were used for the Custom Fuel Dissolution process. Custom processing ceased in 1982. The entire CPP-627 building was deactivated in 1997. Demolition of the building was completed in October 2005.

2.1.4   CPP-640 Headend Process Plant

CPP-640, also known as the HPP, contains approximately 15,000 ft^2 of floor space and was constructed in 1961, with operations beginning in 1963. The facility consists of six shielded cells and a waste collection system, as shown in Figure 2-2. Operations included the electrolytic dissolution process and the Space Nuclear Propulsion Program (Rover), which were discontinued in 1981 and 1984, respectively. The Rover facility provided a headend (fuel dissolution process) system for the reclaiming of uranium for both unirradiated and irradiated Rover fuels for the Atomic Energy Commission Rocket Program. The electrolytic dissolution process was used specifically for the recovery of uranium from fuels with stainless-steel cladding. Cells 2–4 and the shielded mechanical handling cave were used for the Rover process, while the electrolytic dissolution process was performed in Cell 5. The aqueous solution from both operations was then sent to CPP-601 to extract the highly enriched uranium.
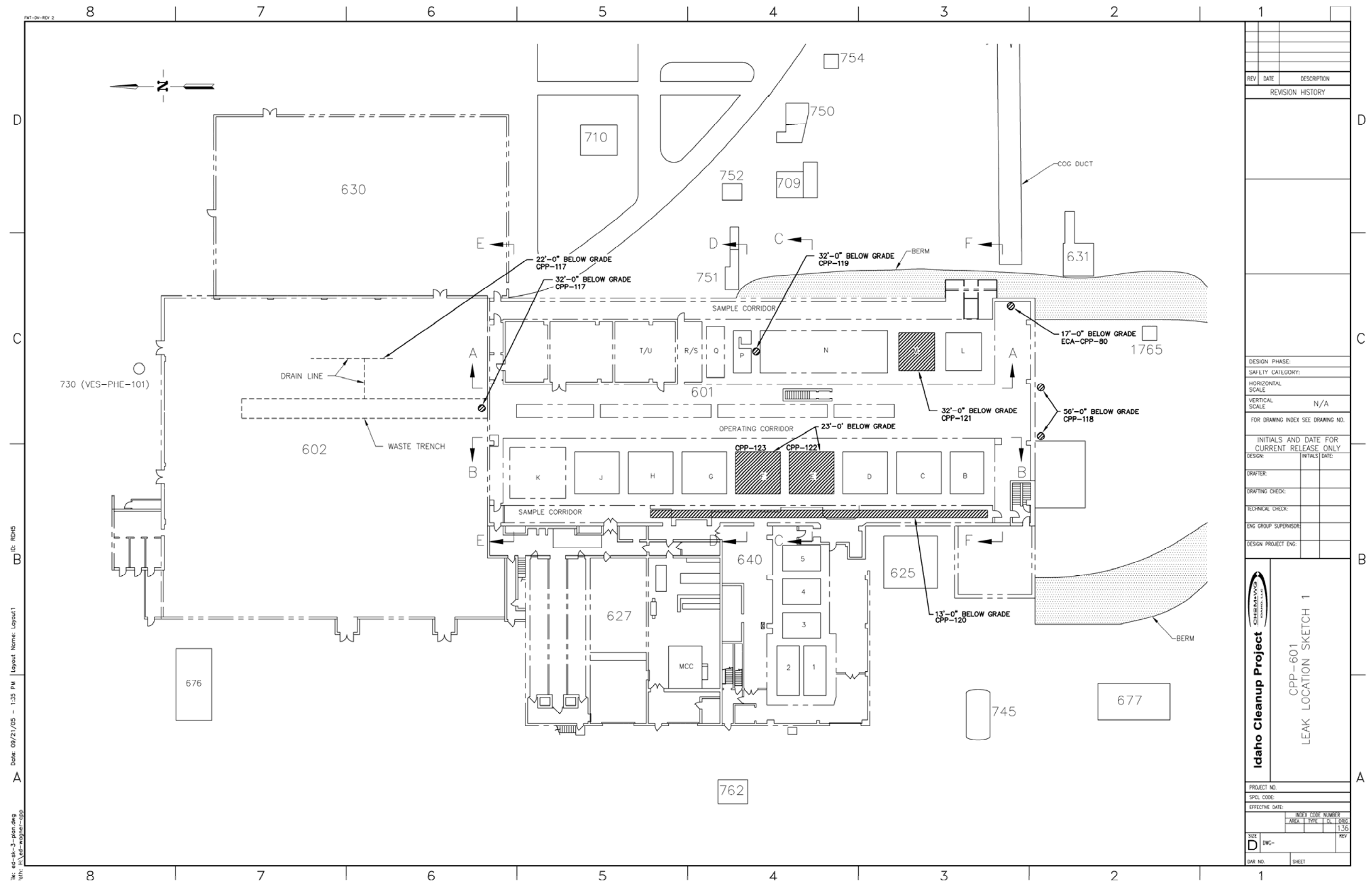
Figure 2-2. Schematic of the Fuel Reprocessing Complex and release locations.

2.2   General Facility Operating History

Any description of FRC function and operation and, hence, of CPP-601 function and operation is complicated by the fact that the plant was continuously evolving. This section provides a summary of major process changes that occurred during the operating lifecycle of CPP-601, the portion of the FRC that performed the majority of fuel reprocessing. The information in this section was derived from the *Process Description and Operating History for the CPP-601/-640/-627 Fuel Reprocessing Complex at the Idaho National Engineering and Environmental Laboratory* (INEEL 1999).

The CPP-601, CPP-602, CPP-627, and CPP-640 facilities (which collectively constitute the FRC) were used to reprocess or support reprocessing spent nuclear fuel at INTEC, formerly referred to as the Idaho Chemical Processing Plant. The CPP-601 facility was constructed in 1953 and included the separation process, dissolution processes such as the aluminum fuel dissolution system and the zirconium fuel dissolution system, chemical makeup and transfer processes, and liquid waste receiving processes. Fuel dissolution processes (also referred to as "headends") were added or modified as needed to match the different types of fuel to be processed. Headends were abandoned when specific types of fuel were exhausted or when superseded by improved processes.

CPP-601 was originally built with three extraction cycles using hexone solvent in P-, Q-, and, S-cells. Batch-type fuel dissolvers were located in A-, C-, and D-cells. A system for extracting barium (Ba) -140 from "green" fuel, called the radioactive lanthanum recovery process, was installed in L-cell. New extraction cycles based on tributylphosphate (TBP)/kerosene solvent were later installed in F-, G-, and H-cells. The F-cell system was designed to operate with the submarine thermal reactor/submarine intermediate reactor dissolution process in E-cell. The G- and H-cell system was designed to operate with the continuous aluminum dissolvers in G-cell. The TBP/kerosene cycles were then used for first-cycle extraction. The P- and Q-cells were modified to serve as second and third cycles (hexone). The extraction cycle in S-cell was abandoned along with the batch dissolvers. The electrolytic dissolver in CPP-640 replaced the submarine intermediate reactor dissolver, and the addition of nuclear poison to the submarine thermal reactor process allowed it to use the G- and H-cell extraction system. The F-cell extraction system and B-cell product collection tanks were then removed. Centrifuges for product clarification were installed in F-cell.

Raffinate collection systems also changed with the extraction cycles. First-cycle raffinates were collected in U-cell (from P-cell and radioactive lanthanum recovery process), E-cell (from F-cell), and G-cell (from G- and H-cells). Second-cycle raffinates were collected in Y-cell (from Q-cell) and U-cell (from P-cell). Third-cycle raffinates were collected in Y-cell (from S-cell and later from Q-cell). Y-cell was reconfigured from four pairs of aqueous raffinate collection tubes and four pairs of hexone collection tubes to a pair of eight-tube raffinate collection banks. The collection banks in U-cell and Y-cell were then connected in series so that the combined second-cycle and third-cycle raffinates could be given two independent sample points to ensure criticality safety. The raffinate evaporators in U- and Y-cells were abandoned as unnecessary.

Carbonate wash solutions from the TBP/kerosene cycles were originally routed to the PEW system, but were later routed through U- and Y-cells like the second-cycle and third-cycle raffinates. Hexone was originally collected in W-cell (from P-cell) and in Y-cell (from Q- and S-cells). Later, all hexone collection was combined in W-cell.

Accountability measurement tanks were installed in M-cell to provide more timely control of nuclear materials. One pair of tanks was inserted into the flow of first-cycle product to the N-cell intercycle storage banks. The second pair was inserted into the flow of third-cycle product to the Z-cell product storage banks.

A density monitor tank, S-116, was installed in S-cell to solve part of the criticality safety weaknesses in the original uranium salvage system located in J-cell. To alleviate further criticality safety weaknesses, a new criticality safety-salvage system was later installed in L- and C-cells. The J-cell was then restricted to the recycle of PEW solutions. S-116 was then reconfigured to serve as a decanter to prevent TBP from reaching the L- and C-cell salvage system. The original final product was liquid uranyl nitrate, which was loaded into bottles in the CPP-602 basement. A fluidized bed denitrator that produced granular solid uranium trioxide replaced that system. The denitrator also was located in the CPP-602 basement.

The U.S. Department of Energy (DOE) ended nuclear fuel reprocessing at CPP-601 in 1992. The facility was no longer needed, making it obsolete for its originally intended mission. The PEW system continued to collect and chemically adjust the low-level and intermediate-level radioactive mixed waste streams that originated from the various sources in the FRC, from the inactive fluorinel dissolution process cell, and from the active laboratories in CPP-602 and CPP-684. The four storage tanks that constitute the PEW system in CPP-601 are VES-WG-100, VES-WG-101, VES-WH-100, and VES-WH-101.

3.   FUEL REPROCESSING COMPLEX PROCESS DESCRIPTION

The INTEC has been in operation since 1954 and was historically a uranium reprocessing facility for defense and research projects and a storage facility for spent nuclear fuel. Irradiated nuclear fuels were reprocessed to recover unused U-235. The fuels processed at INTEC were highly enriched (between 20 and 97% U-235; naturally occurring materials typically have a U-235 concentration <1%). The recovery of uranium from spent nuclear fuel was driven by the economic value of uranium (approximately four times that of gold), worker safety, and national security issues. This section describes the general fuel reprocessing process and the associated handling of waste during the process. Leaks that have been identified during the fuel reprocessing are presented in Sections 4 and 5, following the FRC process description.

The uranium dissolution and extraction process at INTEC includes chemical processing equipment, located in the Fuel Process Building (CPP-601), associated buildings (the Laboratory/Offices Building [CPP-602], the RAF [CPP-627], and the HPP [CPP-640]), and, later, the Fluorinel Dissolution Process and Fuel Storage (FAST) Facility (CPP-666). The recovery of uranium from spent nuclear fuel involved a variety of dissolution and extraction processes tailored to the specific fuel types being processed.

In order to recover uranium in an efficient, continuous, and safe manner, the uranium dissolution and extraction process was divided into relatively small process cells so that portions of the process could be temporarily deactivated and isolated for decontamination and maintenance. The equipment used minimized personnel radiation exposure required for routine maintenance on the system. The process was designed to allow alternative solution routes around failed or deactivated equipment. Redundant equipment was often installed to allow operations to continue during decontamination and maintenance operations. Process equipment was chosen for durability, simplicity, and minimum use of moving parts and seals. This resulted in the use of steam jets and airlifts, as opposed to the use of mechanical pumps in many of the process segments.

Safety concerns associated with radiation exposure and criticality dictated the maximum allowable uranium concentration within a vessel and the geometry of vessels used to handle uranium solutions. Radiation safety and uranium criticality were managed through the use of procedural controls that governed the transfer and sampling of uranium solutions and through the design of the size, shape, and spacing of process units and lines.

Because highly enriched uranium can be used for nuclear weapons, national security issues required an annual inventory of uranium. Therefore, a mass balance of uranium entering and leaving the uranium dissolution and extraction process was performed. Following a dissolution and extraction campaign, immediate, extensive, and repeated flushing was required to complete the mass balance and meet national security requirements.

Effective and efficient decontamination minimized the potential for radioactivity exposure to personnel during hands-on maintenance of the processes. Vessels were selected and designed so that they could be completely drained. Piping within the uranium dissolution and extraction process was pitched to drain to low points and back to vessels in the process. Valves and pumps used in the process were selected with minimum internal volume. In addition, numerous flush lines were installed to allow for the decontamination and flushing of process lines.

3.1   Uranium Dissolution and Extraction Process

The dissolution and extraction process at INTEC was designed to recover uranium from spent nuclear fuels using an acid dissolution followed by a liquid-liquid extraction of the uranium. The uranium was purified through these processes to a solid granular uranium trioxide ready for packaging. Some fuels required applied electrical currents or preliminary combustion of graphite in addition to the acid dissolution process. The following sections describe the basic steps of the uranium dissolution and extraction process as it existed most recently.

3.1.1   Fuel Dissolution

Dissolution of spent nuclear fuel at INTEC was performed in several different locations, or headends, producing dissolver product solutions from which uranium could be extracted using a liquid-liquid extraction process. The dissolution of spent nuclear fuels involved dissolving fuels in a highly concentrated acid solution. This dissolution process was facilitated by the addition of various soluble catalysts and neutron poisons, which created chemically complex uranium solutions. Headends were added or modified as needed to match the different types of fuel to be processed. Headends were abandoned when specific types of fuel were exhausted or when superseded by improved processes.

Custom processing of small lots of specialized fuels was done in the HCL or the MCC in CPP-627. The dissolver product from the custom dissolvers was sent to the uranium salvage system located in the L-cell of CPP-601 for purification and uranium accountability sampling. From the uranium salvage system, the dissolved fuel either was sent to the first-cycle extraction process or was concentrated and sent to intercycle storage for second- and third-cycle processing, depending on the chemistry and radioactivity of the fuel.

Zirconium-clad nuclear fuels were dissolved in the fluorinel dissolution process located in CPP-666 (not part of the FRC). Advanced naval fuels were dissolved using a modified-batch process using hydrofluoric and nitric acids. The acidic dissolver product was transferred to the feed preparation process in CPP-601.

Graphite-clad nuclear fuels from the Space Nuclear Propulsion Program, also known as Rover, were processed in CPP-640. The graphite was combusted in a two-step process, and the ash was then dissolved in hydrofluoric and nitric acids. Stainless-steel-clad nuclear fuels were dissolved using the electrolytic dissolution process located in Cell 5 of CPP-640. This electrolytic process used nitric acid and a direct electric current to dissolve the spent fuel. Dissolver product from both the Rover dissolution process and the electrolytic dissolution process was transferred to the feed preparation process in CPP-601.

Dissolution of zirconium-clad nuclear fuels also was performed in the E-cell of CPP-601 using hydrofluoric acid. Aluminum-clad nuclear fuels were dissolved in the G-cell of CPP-601 using nitric acid. These dissolution processes, located in CPP-601, could be operated separately or jointly in a manner called coprocessing. Coprocessing reduced waste volume by using the dissolved aluminum cladding to complex the fluoride ions in the zirconium dissolver product instead of adding aluminum nitrate as a reagent. Dissolver solutions from both systems were transferred to the feed preparation processes in CPP-601.

3.1.2   Feed Preparation

From the headend processes, the dissolver product solutions were transferred to the feed preparation processes (except custom fuel dissolution solutions) located in the E-, F-, and G-cells of CPP-601. Depending on the fuel type, feed preparation consisted of holding the dissolver product at an elevated temperature to digest residual solids, centrifugation to remove solids, chemical adjustment, complexing, uranium accountability measurement, and storage. Prepared dissolver product solutions were then transferred to the first-cycle extraction process located in the G- and H-cells.

3.1.3   First-Cycle Extraction and Concentration

The first-cycle extraction process was a liquid-liquid extraction that separated the uranium from fission products and cladding salts. The first-cycle extraction process consisted of four pulsed columns and an evaporator. Dissolver product from the headends was fed to the first column (extraction column) where the aqueous solution was put in contact with the organic extractant (5 to 10% solution of TBP in a refined kerosene [n-dodecane, AMSCO, EXXOL, etc.] diluent).

The uranium-bearing organic solution entered the second column, where a high-salt, low-acid solution (scrub solution) was used to remove additional contaminants from the organic phase. This scrub solution was recycled back to the extraction column feed. The uranium-bearing organic solution entered the third column, where a very dilute nitric acid solution was used to strip the uranium back into the aqueous phase. The aqueous uranium solution then entered the fourth column, where it was washed with a stream of pure refined kerosene diluent to remove any residual TBP or TBP degradation products to protect the product evaporators from a buildup of organic residue. The uranium solution was concentrated in the evaporator from between 2 and 5 g U-235 per liter to between 300 and 400 g U-235 per liter. The concentrated uranium solution was transferred to the M-cell for uranium accountability measurement and then to the N-cell for intercycle storage prior to the second- and third-cycle extraction process.

Spent organic and aqueous solutions produced during the first-cycle extraction process were collected and treated and could be recycled or discharged as waste. Organic solvent used in the first-cycle extraction processes was transferred to the TBP/kerosene solvent treatment processes located in the H-, K-, and S-cells.

3.1.3.1   Tributylphosphate/Kerosene Solvent Treatment and Recycle. The first-cycle extraction solvent was treated and recycled in the H-cell by means of three mixer settlers. The first stage used strip solution to capture and return any residual uranium to the fourth column feed in the first-cycle extraction process. The second stage used a sodium carbonate stream to remove impurities from the solvent. This carbonate solution was recycled at a 90% ratio with fresh carbonate. The 10% spent carbonate was collected in the U-cell and sampled in both the U- and the Y-cells for uranium accountability. If both samples did not exceed uranium limits, the solution was transferred to the Tank Farm Facility (TFF); if either sample failed, the solution was recycled through the uranium salvage system. The third-stage mixer settler used strip solution to neutralize residual carbonate. The treated solvent was collected and recycled back to the first column.

Wash solvent (pure diluent) from the fourth column of the first-cycle extraction process was collected, sampled for uranium accountability, and transferred to the K-cell for treatment. Waste solvent was treated in a steam stripper, where residual fission products, transuranic elements, and heavy metals were removed. After 1984, the purified solvent was collected and sent to the NWCF Solvent Storage System (Voluntary Consent Order System INTEC-081) (INEEL 2002) for storage prior to being used as a fuel for the NWCF. Prior to the completion of the NWCF in 1984, the solvent was incinerated in a

solvent burner located near the base of the main stack, which has been subsequently removed. At the end of a processing campaign, spent first-cycle extractant also could be processed through the K-cell and sent to the NWCF for use as fuel.

3.1.4   Uranium Accountability

Aqueous solution from the first column in the first cycle was collected as first-cycle raffinate and sampled for uranium accountability. If the uranium levels were too high, the raffinate could be recycled for uranium recovery. If the uranium level met the specified limits, the raffinate solution was transferred to the TFF as high-level waste.

Concentrated uranium solution from the first-cycle evaporator was transferred to the M-cell for uranium accountability measurement and then to the N-cell for intercycle storage prior to the second- and third-cycle extraction process.

3.1.5   Intercycle Storage

Intercycle storage consisted of six criticality-safe collection banks located in N-cell. The banks were used for storage of first-cycle extraction solution prior to the start of the second- and third-cycle extraction processes.

3.1.6   Second-Cycle and Third-Cycle Extraction and Concentration

When the N-cell intercycle storage was filled, the dissolution and first-cycle extraction processes were shut down, and the second- and third-cycle extraction processes were started. The second- and third-cycle extraction processes are nearly identical and are located in the P- and Q-cells in CPP-601. The processes were operated in series and used liquid-liquid extraction and evaporation to further concentrate and purify the uranium solutions. The extraction processes used hexone as the organic solvent and followed a similar process to that followed in the first cycle. Each extraction process consisted of two columns, the first of which combined the extraction and scrub functions of the first-cycle extraction process. The second column was the stripping column. In each cycle, the columns were followed by an evaporator to concentrate the uranium solution for most efficient processing. The chemistry of the second- and third-cycle extraction processes precluded the need for a wash column. Spent organic (hexone) produced in the second- and third-cycle processes was recycled back into the extraction process. Spent hexone solutions were transferred to the K-, T-, and W-cells for hexone solvent treatment and storage. Aqueous solutions could be recycled back to the process, if they contained recoverable amounts of uranium, or discharged as a waste.

3.1.6.1   Hexone Solvent Treatment/Storage and Recycle. The used hexone from the second- and third-cycle extraction processes was collected in the W-cell for recycle. The hexone was fed to a plate column still in the K-cell. A countercurrent sodium hydroxide wash stream removed impurities from the hexone. The purified hexone vapors were condensed and collected. The T-cell was used to store the purified hexone and to feed it to the extraction columns of the second- and third-cycle extraction processes.

3.1.6.2   Raffinate/Carbonate Collection and Sampling. The aqueous raffinates from the first column of the second- and third-cycle extraction processes were collected in both the U- and Y-cells for uranium accountability and criticality safety. The same tanks were used for collecting and sampling spent carbonate solution from first-cycle solvent treatment. If both samples did not exceed uranium limits, the solution was transferred to the TFF; if either sample failed, the solution was recycled through the uranium salvage system.

3.1.7   Uranium Accountability

The aqueous uranium solutions were transferred from the third-cycle extraction process to the M-cell for uranium accountability. After adequate uranium accountability sampling had been performed, the uranium solutions were transferred to the Z-cell for final product storage prior to denitration.

3.1.8   Final Liquid Product Storage

The final liquid product storage consisted of three criticality-safe storage banks. The banks were used for surge capacity for the denitrator process. The uranium solutions were transferred from the storage banks to the denitration process for solidification and packaging.

3.1.9   Denitration

The heated fluidized bed denitrator, which was operated in the basement (LC area) of CPP-602, converted the liquid uranium solution to a solid, granular uranium trioxide. The solid uranium trioxide was then packaged and stored as the final product of the uranium dissolution and extraction process.

3.1.10   Uranium Salvage

The uranium salvage system was used to recover uranium contained in solutions generated from process upsets, leaks in the process cells, and decontamination solutions from process piping and equipment. The uranium salvage system also collected the dissolver product from the custom fuel dissolution process. Uranium solutions were concentrated and chemically adjusted for return to the extraction cycles. If it was suspected that the uranium content of a solution was too low to be recovered, the solution could be sampled in the L-cell and again in the C-cell. If both samples showed sufficiently low uranium, the solution could be transferred to the PEWE system via the CPP-601 deep tanks.

The J-cell was used for the rare recycle of solution from the deep tanks. The CPP-601 deep tanks were always sampled for uranium accountability prior to discharging the contents to the PEWE system. Only when it was determined, based on analytical results, that there were insufficient levels of uranium for recovery were solutions discharged to the PEWE system. The J-cell evaporator could be used for initial concentration of the recycle solution. This solution would then be transferred to the L-cell by way of the S-cell decanter system, which would detect the presence of any first-cycle organic in the solution and return it to the first-cycle solvent treatment process.

3.1.11   Cold Processes and Decontamination Chemical Makeup and Feed (Process Makeup Area)

The PM area is an unpartitioned area on the top floor of CPP-601, which was used to supply feed and makeup process chemicals to the extraction systems, the solvent treatment systems, and the dissolvers in CPP-601, CPP-627, and CPP-640. The PM area also was used to supply decontamination solutions and bulk chemicals (nitric acid and aluminum nitrate) to the process cells in CPP-601, the Waste Process Building (CPP-604), CPP-627, CPP-640, and CPP-659.

A typical decontamination cycle consisted of a strong caustic solution to break down the resistant oxide layer, followed by corrosive and chelating agents to remove exposed base metal and contaminants. The cycle was completed with a 6 M nitric acid flush to restore the oxide layer. The flushes were sampled for radioisotopes to follow the progress of the chemical decontamination. When the amount of radioisotopes being removed dropped off, another cycle was begun using a different mix of chemicals. The decontamination process could last months, continuing until the activity of the process lines and

vessels was sufficiently reduced for personnel access (INEEL 1999). Decontamination solutions used in the uranium dissolution and extraction process and typical concentrations are identified below:

- Water

- Nitric acid (2 to 6 M)

- Potassium permanganate in 2 M nitric acid

- Turco 4521 (commercial product composed of ammonium oxalate, oxalic acid, amorphous silica, citric acid, and 2-mercaptobenzothiazole)

- Turco 4502 (commercial product composed of potassium hydroxide, potassium permanganate, potassium chromate, and hexavalent chromium)

- Oxalic acid in 0.2 M citric acid

- Sodium hydroxide in 0.15 M tartaric acid

- Turco 4324 (commercial detergent composed of ammonium bicarbonate, sodium hexametaphosphate, octylphenoxypoly [ethyleneoxy]ethanal, noctylphenoxpoly [ethyleneoxy]ethanol, and sodium carboxymethyl cellulose)

- Oxalic acid (1 M)

- Citric acid (0.5 M)

- Oxalic acid in 0.5 M nitric acid

- Ammonium hydroxide (0.5 M)

- Sodium hydroxide in 0.5 g/L EDTA (0.0017 M) solution

- Turco ARR (commercial product used for de-scaling that consists of sodium hydroxide, kerosene, triethanolamine, diethanolamine, and sodium gluconate)

- Sodium fluoride

- Hydrofluoric acid in 2 M nitric acid.

3.2 Waste Discharges

All units associated with the uranium dissolution and extraction process had a route (either direct or indirect) to one of the INTEC waste systems (CPP-601 deep tanks, CPP-640 waste collection tanks, or the TFF). Process liquids, which do not include evaporator overheads, were not discharged to the waste system unless samples showed that there was insufficient uranium for recovery. The following sections discuss the waste systems associated with the uranium dissolution and extraction process in CPP-601.

3.2.1 CPP-601 Process Equipment Waste Collection System (CPP-601 Deep Tanks)

The CPP-601 PEW collection system collects low-level and intermediate-level liquid waste from processes within CPP-601. Liquids are transferred to four 4,500-gal deep tanks (VES-WG-100, VES-WG-101, VES-WH-100, and VES-WH-101) located in the WG- and WH-cells in CPP-601. The deep tanks received waste from drains and sumps or transfers from processes in CPP-601, CPP-602, CPP-627, CPP-640, CPP-666, and CPP-684 RAL. Although most process sources of liquid waste for the deep tanks are currently inactive, the deep tanks still have the ability to receive liquid waste from the laboratories in CPP-602 (active), CPP-684 (RAL active), CPP-666 Fuel Dissolution Process dissolution cell, and water infiltration collected in sumps throughout CPP-601. Solutions collected in the deep tanks are normally transferred to the PEWE system. However, the capability exists to transfer solutions directly to the tank farm. Rarely, during fuel reprocessing, solutions were recycled from the deep tanks back to the uranium dissolution and extraction process for uranium recovery. The deep tanks also are configured such that they could also be discharged to the TFF.

Twenty-two separation units discharged directly to the CPP-601 PEWE collection system either on a continuous or routine basis during the operation of the uranium dissolution and extraction process. Units that continuously discharged to the CPP-601 PEWE collection system did so as a normal part of their operation. These continuously discharging separation units discharged solution to the process, as well as the CPP-601 PEWE collection system, and were operated as process units. Other process units infrequently discharged to the CPP-601 PEWE collection system during operation. Solutions discharged to the CPP-601 PEWE collection system were determined to be a waste only when sampling results indicated that the uranium concentration was not sufficient for recovery. Therefore, the solutions were not classified as a waste until they were discharged into the CPP-601 PEWE collection system. In addition to discharges to the CPP-601 PEWE collection system during operations, decontamination solutions also could be directly discharged to the CPP-601 PEWE collection system.

As originally conceived, there were to be two waste collection systems: a PEW system to collect waste generated during processing and a cell floor drain system to collect floor drainage and decontamination solutions. The separation proved to be impractical and within a short time (before 1962) the systems were combined into one PEW collection system.

The main collection headers were routed through a waste trench beneath CPP-601. The process cells, service corridor, waste trench, and tank vaults were constructed with stainless-steel floor liners, which extended part way up the walls. The floors were sloped to leak detection sumps, which contained transfer jets for returning solution to the vessels. Process cell floors were constructed with drains to the old cell floor drain header. In many cases, the drains posed a criticality risk; therefore, those drains were capped and replaced with jets from the sumps. During operation, solutions that collected on the floor sumps were transferred to uranium salvage. During decontamination and maintenance activities, floor sumps and drains transferred solutions to the PEW tanks.

The original lines connecting the process cells with the collection headers in the waste trench were directly buried in the soil under the building without secondary containment. These lines were replaced in 1990–1991, with new collection headers routed through the service corridor and vent tunnels. Where necessary, floors were graded for drainage and provided with leak-detection sumps and compatible liners. The old lines were capped. The lines were originally pitched for gravity drainage and were considered empty. These lines were not flushed, as leaks were known to exist in portions of the system. Attempts to flush the lines could have introduced additional waste to the soil. The leaks in the piping referenced above are discussed in Sections 4 and 5 of this report.

4.   KNOWN FUEL REPROCESSING COMPLEX SYSTEM LEAKS

Prior to this report, the only documented release known to have occurred within the FRC system was the CPP-80 release that was included in the *Comprehensive RI/FS for the Idaho Chemical Processing Plant OU 3-13 at the INEEL—Part B, FS Report (Final)* (DOE-ID 1997). Further details on the release are provided in the following subsections in order that this document presents all known releases and associated radiological activity.

4.1   CPP-80, CPP-601 Vent Tunnel Drain Leak

The release location is below the South Vent Tunnel drain in CPP-601 and along the route of the cast-iron line (4" SK-1075T). The release was believed to have occurred between 1982 and 1989.

4.1.1   Description of the Release

On July 20, 1989, it was discovered that the 4-in. SK-1075 cast-iron line that served as a floor drain in the CPP-601 South Vent Tunnel had failed (Figures 2-2, 4-1, and 5-1). The 4-in. line penetrated the 2-ft-thick concrete vent tunnel floor at 12 ft below ground level (bgl) and was installed down to a depth of 34 ft bgl, where the line construction material changed to stainless steel before entering the deep PEW tank vault. The failure was attributed to corrosive solutions from the VT-300 vessel off-gas (VOG) condenser, which condensed all gaseous vapors from the CPP-601 fuel processing equipment. The condensate, containing radioactive and hazardous contaminants, was released to the soil between the South Vent Tunnel floor and the point where the cast-iron pipe meets the stainless-steel line.

4.1.2   Background of System Configuration and Leak

The VT-300 VOG condenser received and condensed all gaseous vapors from the CPP-601 fuel processing equipment, except the active dissolvers, whose off-gas was routed to one of the dissolver off-gas (DOG) systems. The condenser removed moisture from the off-gas waste stream before the gas was piped to the CPP-604 VOG treatment system for contaminant removal. The processing equipment included the denitrator vessel, solvent extraction columns, storage vessels, and exhaust from airlift and steam jet operations. The off-gas from these devices originates from vessels that contained all types of process solutions, including dissolver product, raffinates, and extraction solvents.

In the 1982–1983 timeframe, the drain from VT-300 was moved from the High Vacuum Room drain line (3" 11641C), which consisted of stainless steel, to the South Vent Tunnel and connected into the drain on 4" SK-1075T, which was constructed of cast iron. The reason for moving the line is not known, but it is believed to be associated with the FAST lines tie-in at CPP-601 that brought in dissolver solutions from the FAST facility for uranium recovery.

Corrosion and subsequent breaching of the line are believed to have occurred sometime after 1982, after the VT-300 condensate was rerouted from its original stainless-steel line to the SK-0175T cast-iron line. The release began potentially at that time and continued until 1989 when the failure was discovered as a part of a systematic review of plant drain lines. As a part of the leak investigation, water was added to the VT-300 condenser loop seal to determine if it would be received in the WG/WH deep tanks. The test was inconclusive, and the investigation proceeded to the actual drain in the South Vent Tunnel. Water was added directly to the floor drain, but the water failed to enter the deep tanks. The condensate drain line was then visually inspected and found to have failed. Sampling of the concrete/debris at the drain location was attempted but was unsuccessful. Radiation readings from the drain hole cavity were 1.5 R/hr.
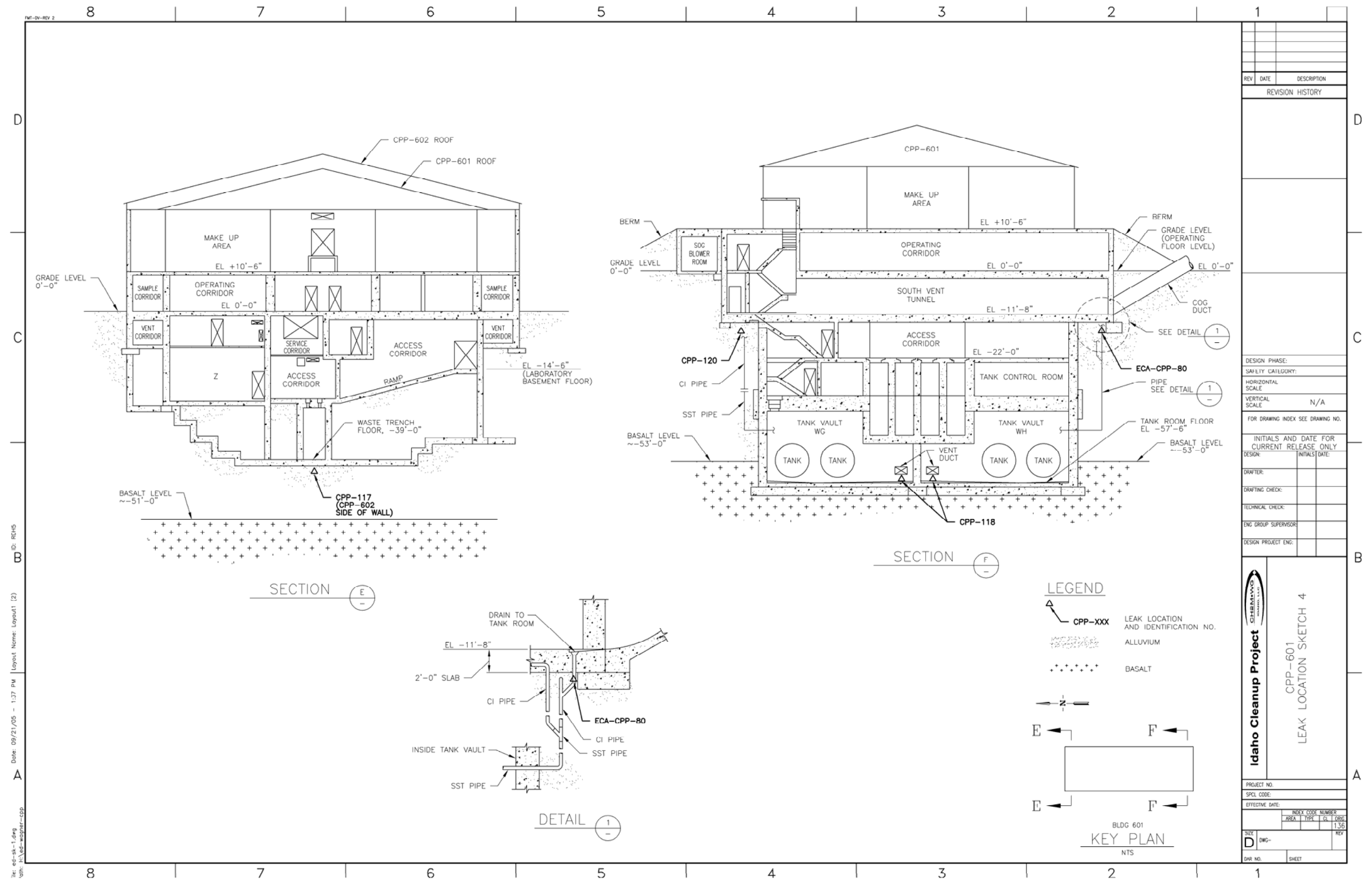
Figure 4-1. Cross-section view of CPP-601, from south, with release locations identified.

The use of the drain line was discontinued; pipes were capped and the cavity left in the concrete floor where the floor drain was located was filled with grout. The drain line from the VT-300 tank was rerouted to a service corridor floor drain that drained to the WG/WH deep tanks.

Further investigation indicated that a drain line from a DOG line intersected the vent tunnel's floor drain at a point deeper underground. When water was added to the DOG line, it reached the WG/WH deep tanks, indicating that this line connected to the VT-300 drain line, but at a point below the failed section of line.

4.1.2.1 **Waste Source Term.** Determining a source term for the CPP-80 (VT-300) leak is difficult to estimate. Because sample ports were not installed in the condensate drain line, no samples of the VT-300 condensate were collected for analysis, and, apparently, process flow sheets were not generated for the system. The waste would be primarily off-gas condensate; thus, sample results from condensate samples collected in the CPP-649 off-gas system can provide a basis for determining a waste source term. The CPP-601 off-gas system vents into the CPP-649 off-gas system; therefore, at the first glance, the samples would represent activities of the condensate that leaked into the soil. However, the CPP-649 off-gas system included off-gas from the Waste Calcining Facility, the NWCF, and the tank farm. To help determine contaminant levels, various data sets were considered and factored into the development of a reasonable source term. A description of how the source term was determined is presented in Appendix A. The work resulted in the following estimated radionuclide activities:

- Cs-137—1.33×10^{-7} Ci/L

- I-129—1.33×10^{-10} Ci/L

- Sr-90—1.33×10^{-7} Ci/L

- H-3—1.33×10^{-6} Ci/L

- Tc-99—1.33×10^{-9} Ci/L.

Table 4-1. Radionuclides of concern released at the CPP-80 site.

Radionuclide	Radioactivity per L (Based on Estimated Off-Gas Concentrations) (Ci/L)	Radioactivity Released Assuming 399,460 L of Waste Released (105,540 gal)
Cs-137	1.33×10^{-7} Ci/L	5.29×10^{-2} Ci
Sr-90	1.33×10^{-7} Ci/L	5.29×10^{-2} Ci
H-3	1.33×10^{-6} Ci/L	5.29×10^{-1} Ci
I-129	1.33×10^{-10} Ci/L	5.29×10^{-5} Ci
Tc-99	1.33×10^{-9} Ci/L	5.29×10^{-4} Ci
	Total	6.35×10^{-1} Ci

4.1.2.2 **Waste Volume Leaked.** The volume of off-gas condensate released in the soil below the floor drain's failure point is uncertain. Depending on the FRC activity, the amount of condensate originating from the VT-300 vessel would vary. During the time the cast-iron drain was used, facility operations were producing VT-300 off-gas condensate at flow rates of 1 to 5 L/hr during normal operations, depending on the number of plant operations in progress, and up to 150 L/hr during steam decontamination activities. Because of the configuration of the drain system and its location, it was not possible to measure actual flow rates. Therefore, past estimates used a continuous flow rate of 6 L/hr as a reasonable estimate to bound the volume of condensate released. Assuming that the cast-iron line failed on January 1, 1982, and continued through July 20, 1989 (7.6 years or 2,774 days or 66,576 hours), the total liquid lost would equal 399,460 L or 105,540 gal. It is possible that this number is conservative because some of the liquid might have moved through the original pipe path, ending up in the deep tanks. However, the assumption will be made that none of the liquid made its way into the WH/WG deep tanks.

4.1.2.3 **Summary.** The CPP-80 release was the result of not ensuring material compatibility before rerouting the VT-300 off-gas condensate drain line. The acidic condensate water containing radioactive fission products was redirected from a stainless-steel line to a cast-iron drain line as a matter of convenience. Both drain lines routed the condensate to the WG/WH deep tanks. The cast-iron drain was not compatible with the acidic condensate and likely failed soon after it began receiving the fluid. The liquid then migrated into the soil beneath the South Vent Tunnel floor. Using an average flow rate of 6 L/hr for the condensate, a total of 399,456 L might have been released to the soil.

The condensate did contain moderate concentrations of radionuclides, evidenced by the 1.5-R/hr readings around the floor drain.

4.1.3 Cleanup

No cleanup activities have been completed for this release because of its location under the South Vent Tunnel wall. Immediately after the discovery of the corroded drain line, the condensate drain line was rerouted, thereby ending the leakage into the soil. The drain hole through the concrete floor was grouted to provide shielding from contamination located beneath the slab.

4.1.4 Previous Investigations

No soil samples were collected from the release site to help characterize the release because of difficulties in obtaining samples. Radiation readings from the drain hole cavity were measured at 1.5 R/hr upon discovery of the leak. Sampling of the VT-300 off-gas condensate was proposed as an action item to characterize the condensate after the drain line reroute was completed. However, because of the loop seal configuration and radiological exposure concerns, a sample port was never installed. Additional piping associated with the drain line was tested and found to be intact.

4.1.5 Contamination Remaining in Alluvium

The release occurred at a depth of approximately 15 to 20 ft bgl in the surface alluvium found at INTEC. The depth to the basalt/alluvium contact beneath the release is estimated to be 50 ft bgl, leaving 30 to 35 ft of alluvium for the leak to saturate before entering the basalt. Considering the vertical routing of the cast-iron line and because the release occurred between the vent tunnel floor and the point at which the line joined the stainless-steel line, it is speculated that the contamination dispersed in a conical pattern was limited to the west by the deep tank foundation wall that rests on basalt.

4.1.5.1 **Extent of Contamination.** Assuming that approximately 400,000 L were released during the 7.6-year leakage period and the soil was completely saturated and had a porosity of 25%, the estimated volume of soil would be 1,600,000 L or 56,000 ft^3. This represents a block of soil 38 × 38 × 38 ft. However, considering that the contamination most likely spread in a downward conical shape (disregarding the cement foundation wall to the west) at a 30-degree angle off vertical to a depth of 50 ft bgl, the volume of contaminated soil would total approximately 15,000 ft^3. This volume represents about 27% of the possible contaminated soil volume, suggesting that a large portion of the released condensate might have entered the basalt after having passed through saturated alluvium.

4.1.5.2 **Remaining Curies.** The radionuclides of concern released totaled 0.635 Ci, as presented above. The only removal of source material has been radioactive decay. The majority of contamination is from Cs-137 and Sr-90, with half-lives of 30 years and 28.8 years, respectively. Therefore, approximately 70% of the originally released activity would still be present, approximately 0.445 Ci.

5.   NEWLY IDENTIFIED HISTORICAL RELEASES

The newly identified releases are based on personal knowledge of workers who were familiar with the FRC systems and had first-hand experience in the reprocessing of spent nuclear fuel. Each release is discussed in further detail below.

5.1   CPP-117, CPP-602 Waste Collection System

5.1.1   Description of Release

The contamination associated with the CPP-602 waste collection system was the result of liquid leaks falling onto the CPP-602 basement floor (pipe leakage, careless cleaning, fire sprinkler water, and rainwater infiltration into the building) or any releases that would have been associated with the CPP-602 laboratory PEW collection piping, feeding into the CPP-602 waste trench. The effects of the releases were discovered in 1989, some time after the releases occurred.

5.1.1.1   ***Background of System Configuration and Leak.*** The LC-107 sump is located at the south end of the CPP-602 waste trench, and its south wall is the common wall between CPP-602 and CPP-601. The sump has a cross section of 3.0×5.0 ft. The depth of the sump floor at the time of construction was even with the CPP-601 waste trench, approximately 39 ft bgl. The sump was constructed using nonwatertight cold joints at the CPP-601 wall and at the bottom. The sump was tied to the CPP-602 waste trench, which directed any liquid leaks from the PEW collection piping into the sump at approximately 22 ft bgl, giving a sump depth of approximately 17 ft. The CPP-602 basement floor, located 14 ft bgl, also drains any liquids released onto the basement floor into the sump. Liquids that accumulated in the sump were not routinely managed and were allowed to drain into the soil beneath the sump floor.

The CPP-602 waste trench originally housed two stainless-steel collection headers that collected PEW from various laboratory drains. One of the headers collected waste from a stainless-steel laboratory drain system. The lab hood sink drains were typically plumbed with stainless steel, compatible with the expected waste generated in the laboratories. The second header collected laboratory waste originating from two cast-iron floor drains and five cast-iron bench sink drains. The cast-iron drain system was shaped in a "Tee" configuration with the drains connecting to two branch headers, which were connected to a single line leading to the CPP-602 waste trench header. The drain line systems were buried in the soil below the CPP-602 basement floor, exited the soil through the waste trench cement wall into the CPP-602 waste trench, and connected to the collection headers (Figures 2-2, 4-1, and 5-1). At some point, one of the stainless-steel headers was taken out of service, and all the floor drains were rerouted to the other stainless-steel header. Generally, hot waste was disposed of through the stainless-steel drains located in sinks within the sample hoods and experiment areas. Occasional floor spills were most likely washed down the floor drains and bench sinks, causing the cast-iron piping to corrode because of the acidic nature of the laboratory solutions.

The release was discovered in 1989 and was based on two facts. First, the concrete bottom of the sump had deteriorated to the point of having a consistency of damp sand. Second, the main CPP-602 waste trench contained an etched "high-water" line, approximately 1/16 in. deep that extended more than half the length of the trench. The deterioration of the sump bottom and concrete etching were due to the acidic nature of one or more of the releases that accumulated in the sump/trench drain system. An investigation conducted after the leak discovery found no records or personnel memories of any large volume releases. The timeframe for the releases is any time between 1953 and 1989.
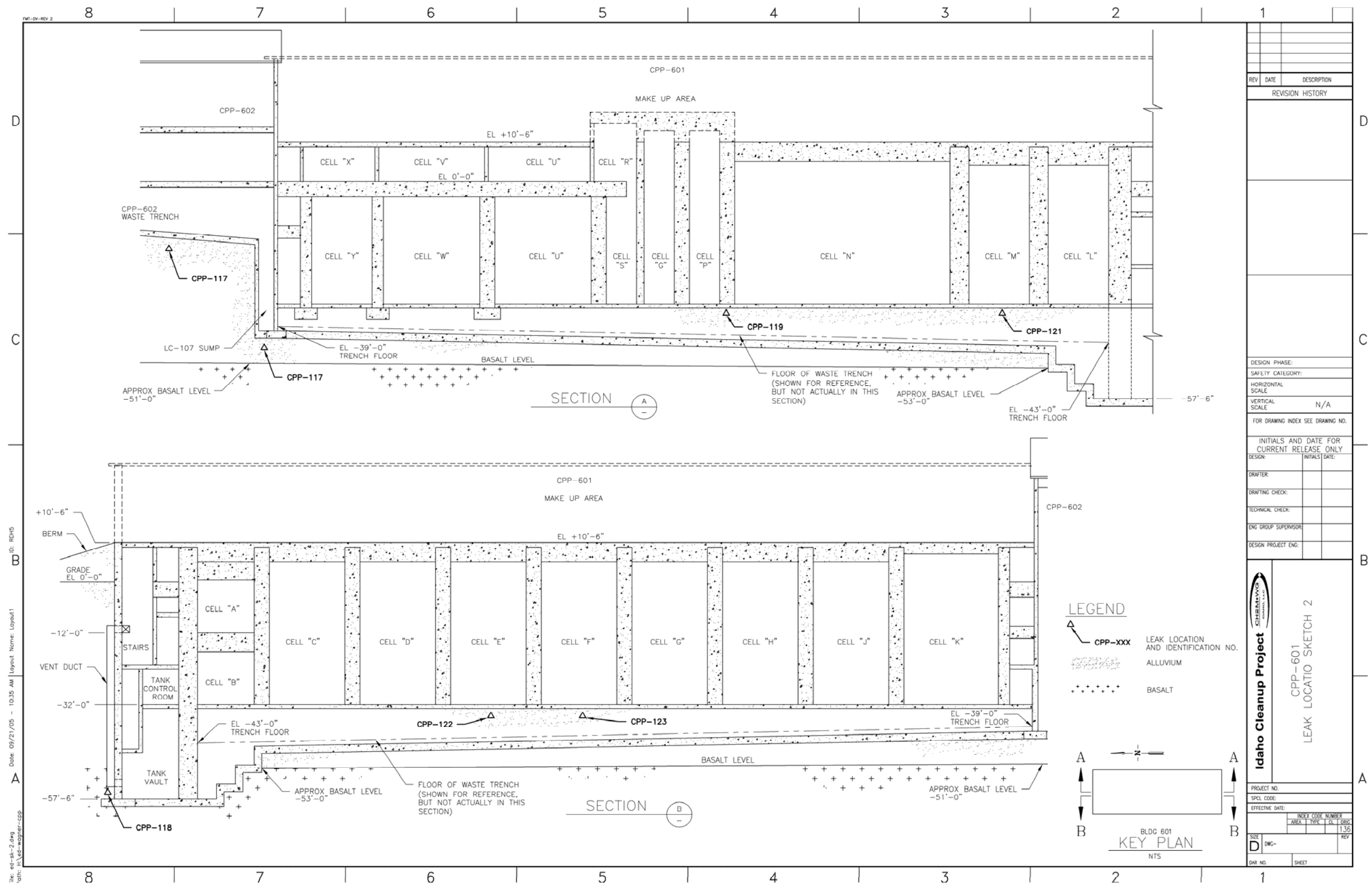
Figure 5-1. Cross-section of CPP-601; top is from middle facing east, bottom is from middle facing west. Release locations are identified.

At the time of the leak investigation, the loop seal in the PEW collection header had a small leak. The piping configuration showed that the loop seal had been replaced at least once previously. In addition, stains near the entry of the cast-iron drain line into the waste trench indicated leakage had occurred from the cast-iron floor drains. Any leakage ended up in the trench and eventually in the sump if sufficient volumes of liquid were present to mobilize the leakage.

It should be noted that the CPP-602 waste trench sump was previously designated CPP-86. Soil samples collected beneath the sump floor were found to contain detectable concentrations of metals but below regulatory action levels. Samples for radiological analyses were not collected. While the CPP-86 site originally focused on the LC-107 sump, the CPP-117 site encompassed the entire CPP-602 waste trench and the buried lines that feed into it. In addition, the description of the CPP-602 waste trench system in this section is more accurate and detailed than the system description provided in the CPP-86 Track 1 investigation document.

5.1.1.2   *Waste Source Term.* The source of contamination that entered the CPP-602 waste trench and LC-107 sump originated from the CPP-602 laboratories. The laboratories generated PEW that was routed through the WG/WH deep tanks. During active laboratory operation, it was estimated that 80% of the sample material generated during testing was recycled back to process uranium recovery. The remaining 20% was discharged into the PEW collection piping. This practice helped minimize the amount of waste generated and reduced the possibility of waste going critical because of increased uranium concentrations (EDF-4366).

The waste was acidic (pH around 0), metal-bearing, and contained typical fission products. Because there were documented small leaks and indications of past leakage of piping in the waste trench, it is reasonable to assume that some of the liquids that collected in the sump had radiological and chemical constituents, including constituents associated with laboratory operations. Radiation readings along the CPP-602 waste trench were relatively low, with readings of 10 mR/hour or less and sump readings of 20 mR/hr or less. These readings support the hypothesis that only minor amounts of laboratory waste were released and that the liquids released had low levels of radioactive constituents. Because laboratory waste solutions sent to the PEW collection system were over time, likely highly variable in terms of radioactivity levels, a typical PEW tank composition was selected from Table 1, "Chemical Composition of PEW Evaporator Feed Solutions and Evaporator Bottoms," in Appendix B to represent the waste source term. The typical PEW had the following activities:

- Cs-137—100 μCi/L

- Sr-90—100 μCi/L

- H-3—1.0 μCi/L

- I-129—1.6 × 10^{-3} μCi/L

- Tc-99—1.36 × 10^{-2} μCi/L.

There were no analytical results for Tc-99. To establish a Tc-99 activity, the Cs-137-to-Tc-99 ratio in a 5-year-old, zirconium-clad fuel was used based on an ORIGIN2 (Croff 1980) model simulation. The ratio is based on the original distribution of fission products created during fuel rod usage. The amount of Tc-99 created is proportional to the Cs-137 activity. If a Cs-137 activity and fuel rod age are known, a corresponding Tc-99 activity can be determined. The assumption made in this case is that the Cs-137/Tc-99 ratio holds for various waste streams that have not undergone thermal treatments.

The waste source term selected for the leaks in the cast-iron floor drain system should represent what was most likely washed down the drain system. This drain system likely did not receive contaminated solutions unless samples were inadvertently spilled or dropped on the floor, requiring decontamination action. To establish a reasonable source term, it can be assumed that samples dropped or spilled on the floor consisted of a typical laboratory PEW.

5.1.1.3 **Waste Volume Released.** Two distinct leak mechanisms exist with the CPP-117 leak site releasing contamination to the subsurface. The first is the known leaking loop seal on the PEW waste trench header. The second is the possible corrosion of the cast-iron laboratory floor drain system where wash solutions might have been lost to the soil as a result of drainpipe failure. The evidence of the second release mechanism is indirect. Staining where the cast-iron pipe penetrated the trench wall indicated that liquid from some source was draining back into the trench. In addition, the underground cast-iron lines were replaced for unknown reasons other than Resource Conservation and Recovery Act (RCRA) compliance (42 USC § 6901 et seq.). The most likely remaining reason is that one or more of the lines had failed. The volume of waste lost to the subsurface for either leak mechanism is uncertain.

A volume of liquid that once filled the trench/sump system can be calculated using the geometry of the LC-107 vault and the CPP-602 waste trench knowing the height of the liquid based on the etched concrete. The sump and waste trench geometry produced a volume of 570 ft^3 (3,920 gal) using a sump geometry of $3 \times 5 \times 17$ ft and trench geometry of $5 \times 1.5 \times 72$ ft. This volume of water was not likely from a failed PEW collection system, as the CPP-602 laboratories did not produce that volume of waste in a relatively short time. Based on personnel interviews, this volume of water was probably the result of a fire line leak within the CPP-602 building that released clean water or an overflow of service waste from the adjacent LC-104 sump. No line-break records were found, suggesting that a fire line leak might have occurred over a longer period of time before repairs were completed. Overflow of service waste required the failure of redundant pumps, but such failures have historically occurred, creating a condition where service waste could accumulate in sufficient quantities to spill over into the adjacent sump. Contamination from small leaks of acidic waste from the PEW collection system within CPP-602 undoubtedly mixed with the water, lowering the pH of the standing water, causing etching of the unlined waste trench walls.

Assuming that the PEW loop seal dripped continuously for a period of 5 years at a rate of 1 mL/minute, the total volume released would equal 2,630 L or nearly 700 gal. The volume of waste released through the corroded floor drains is more problematic. Assuming that a 1-L sample is spilled or dropped in each individual lab per month over a period of 35 years and six active basement labs using the cast-iron drain system, a total of 6 L of laboratory waste would be lost down the cast-iron drain system per month or 72 L of waste per year. Over a period of 35 years, a total volume would be 2,520 L or 666 gal of actual PEW. Further assuming that any spills would be decontaminated, the addition of clean water would increase the waste volume by 10 to 20 times but would not add significantly more contamination.

5.1.1.4 **Summary.** The leakage associated with the CPP-117, CPP-602 laboratory PEW drain system consists of two components. The first is leakage of the PEW header loop seal where dripping of laboratory PEW was observed in the CPP-602 waste trench. The second source of leakage was the corrosion and failure of the cast-iron laboratory floor drain system, which allowed laboratory waste spilled on the floor to be washed down the drains and into the soil after any of the drain lines corroded and failed. Evidence that a large volume of liquid (over 4,000 gal) accumulated in the LC-107 sump and CPP-602 waste trench exists, but low radiation levels throughout the trench and sump suggest that the source of liquid for this release was clean. It is not certain how the water left the trench. Review of records and personnel interviews did not indicate that the released liquid was pumped from the sump.

Therefore, it was conservatively assumed that the liquid drained through the cold joints of the LC-107 sump.

An assumed total of 1,366 gal of PEW was released having a typical PEW profile. Based on what is currently known about the release, this volume is reasonably conservative to bound this release in terms of total curies.

Using the volume released and the activity levels for typical PEW provides a reasonable estimate of radionuclides released to the subsurface. These values are presented in Table 5-1.

5.1.2   Cleanup

No cleanup activities have been completed for these two releases because of their location under the CPP-602 laboratory building. To mitigate the problem of the corroded cast-iron floor drain system, the old drain system was replaced with direct-buried, stainless-steel lines in the mid-1980s. These new lines were abandoned in place and replaced with Hazardous Waste Management Act (HWMA) -compliant lines in 1991 (HWMA 1983).

5.1.3   Previous Investigations

No major investigative work has been completed on the CPP-602 waste trench/sump system. Inspection of the sump determined that the former concrete floor had deteriorated to a consistency of damp sand, and radiation levels were relatively low (<10 to 20 mR/hr). No samples were collected for analysis.

5.1.4   Contamination Remaining in the Alluvium

The amount of liquid likely moving the contamination into the alluvium in the vicinity of the cast-iron floor drain system is assumed to total around 7,370 gal (670 gal of liquid waste and 6,700 gal clean wash water). Assuming that approximately 7,370 gal of contaminated water was released and the soil was completely saturated and has a porosity of 25%, the estimated volume of soil impacted would equal 3,940 ft^3. This represents a block of soil $15.8 \times 15.8 \times 15.8$ ft.

Table 5-1. Radionuclides of concern released at the CPP-117 site.

Radionuclide	Radioactivity per L (Based on Typical Process Equipment Waste Concentrations) (μCi/L)	Radioactivity Released Assuming 5,170 L of Waste Released (1,366 gal)
Cs-137	100	0.517 Ci
Sr-90	100	0.517 Ci
H-3	1.0	5.17×10^{-3} Ci
I-129	1.60×10^{-3}	8.27×10^{-6} Ci
Tc-99	1.36×10^{-2} [a]	7.03×10^{-5} Ci
	Total	1.039 Ci

a. Activity is based on the Cs-137-to-Tc-99 ratio for 5-year-old fuel using the ORIGEN2 (Croff 1980) computer simulation results for the radionuclide products and decay of a nuclear fuel.

The amount of liquid passing through the floor of the LC-107 sump was approximately 4,700 gal (700 gal of laboratory waste and 4,000 gal of clean fire system water). Assuming that approximately 4,700 gal of contaminated water were released and the soil was completely saturated and has a porosity of 25%, the estimated volume of soil impacted would equal 2,510 ft^3. This represents a block of soil 13.6 × 13.6 × 13.6 ft.

5.1.4.1 *Extent of Contamination.* The CPP-602 laboratory cast-iron floor drain system's leakage likely is spread along approximately 70 ft of piping, acting as a line source. Assuming equal distribution along the line and a 30-degree angle (off vertical) of lateral spread, the zone of contamination would be contained in the alluvium to a depth of 10 ft below the drain line, approximately 32 ft bgl.

The LC-sump leakage would act as a point source, creating a conical-shaped contaminated soil body below the point of release at 42 ft bgl. Because the depth to basalt is approximately 52 ft bgl, roughly one-third of the liquid would be retained in the alluvium while the remainder would pass into the basalt.

5.1.4.2 *Remaining Curies.* The number of curies released at the CPP-117 release site totals slightly more than 1.39 Ci, with Cs-137 and Sr-90 making up most of the activity.

5.1.5 Uncertainties/Data Gaps

The uncertainty associated with the two CPP-117 releases lies with the volumes released and the activity of the waste. The actual volumes released are not known and cannot be calculated using process knowledge. The activity levels of the waste generated in the laboratories likely fluctuated on a daily basis. The use of the PEW typical radioactivity levels for the radionuclides of concern appears to be a reasonable approximation. If release volumes were an order of magnitude higher, the corresponding radioactivity released would be 10.4 Ci, a relatively small amount in comparison to other releases at INTEC.

5.2 CPP-118, WG/WH Vaults' Ventilation Outlet Ducts

The CPP-118 site (see Figures 2-2, 4-1, and 5-2) is located at the south end of the CPP-601 building, near the building centerline, and located near the bottom of the PEW tank vaults (53 ft bgl, 63 ft below the level of the access ramp). The material there is fractured basalt and compacted fill.

5.2.1 Description of Release

The CPP-118 site was contaminated at some time during the early 1980s, when the PEW tank vaults were twice flooded with solution to a sufficient depth to allow the liquid to enter the unlined concrete ventilation ducts serving the tank vaults. These ducts have a rectangular internal cross section of 12 × 15 in. and entered the vaults through an 8- × 12-in. grating located 12 in. above the top of the sump. Some amount of liquid probably entered the soil, because the ducts, which are cast onto the outside of the south wall of the tank vaults with cold concrete joins, were not designed to contain liquid. At the time of the flooding incidents, the possibility of leakage from the ventilation ducts was not recognized.

5.2.1.1 *Background of System Configuration and Leak.* The INTEC PEW system collects low- and intermediate-level waste from numerous sources for transfer to the PEWE system in CPP-604. The PEW collection system was described in Section 3.2. The sources of this release and chemical constituents are not precisely known but were materials that normally would have been routed to the WG/WH deep tanks. At least four sources of leakage are known.

Figure 5-2. Cross-section of CPP-601 looking north, with release locations identified.

The first, long-term flooding of the vaults, had at least three sources. One was failure of the WH tank man-way gaskets that allowed collected PEW solution to leak out of tanks when they were filled to near capacity. The second was leakage at one of the selection valves located in the main PEW collection headers in the CPP-601 waste trench. The third source was weld failure in the PEW lines from some cell floor drains, primarily G-cell, at their junction with the headers in the waste trench. The waste trench has a stainless-steel liner that directs leakage to a sump located at the low point in the south end of the trench. The waste collection headers pass through the concrete walls of the tank vaults via 8-in., stainless-steel sleeves located just (<1/2 in.) above the floor liner of the waste trench.

Leakage from the sources in the waste trench was not detected early as a result of an improper set point for the waste trench sump alarm. The waste trench sump, SU-WT-101, was later found to be only 10 in. deep compared to the 15- to 18-in. depth of the usual cell and vault sumps. The sump alarm was set to the same value as a standard sump, but liquid filling the sump was not capable of reaching the alarm set point. The sump overfilled without triggering an alarm, and the rest of the solution then passed onto the WG/WH tank vaults through the openings around the PEW collection headers.

The vault flood was not recognized for several months, because the vault sump-level readings were erroneously attributed to instrument problems. The operators assumed that they were dealing with a normal small leak, in which case the sump steam jet would quickly empty the 1-gal sump. When the sump level did not drop, they assumed that either the instrument was wrong or the jet was plugged. It was usually not possible to see the small addition of the sump contents to the large tank volume, unless the jet was operated for a long time with a significant volume of liquid on the vault floor. That is what eventually led to the discovery of the flooded vaults.

The second incident involved fire water, when a failed fire water line in CPP-627 flooded the PEW tanks via open floor drains. The tanks overfilled and released a mixture of water and PEW solution that was already in the tanks to the vault floors. This solution was returned to the tanks via the sump jets as soon as the broken line was isolated and enough tank space could be made available by transfer to the PEWE system. The PEW solution (acidic [pH<0]), aqueous waste with RCRA metals, listed waste codes, and average to low activities of fission products were derived from process streams and decontamination activities. This flooding incident was of short duration (a few days).

The leakage from the floor drain piping in the waste trench might have been an ongoing problem since the 1960s that was not discovered until 1989. The man-way gasket leakage was first noticed in the early 1980s and was repaired along with the leaking valve in 1985. The high-volume leak at the valve that resulted in the vault flood occurred during 1983–1984, as did the fire water flood. Solutions were in contact with vent ducting for 3 to 4 months or more.

5.2.1.2   *Waste Source Term.* There are no analytical data for the solution that might have leaked from the ventilation ducts at the CPP-118 site. However, the activity in the waste can be estimated from the historical analyses of typical PEW solutions of the time (EDF-5441). Using the average contamination activities for the PEW from samples collected from the deep tanks in the early 1980s, a reasonable estimate of activity released can be determined. The estimate will be conservative for the second flooding incident because of the approximately 10 to 1 dilution of the waste with water. The typical PEW had the following activities:

- Cs-137—100 μCi/L

- Sr-90—100 μCi/L

- H-3—1.0 μCi/L

- I-129—1.6×10^{-3} µCi/L

- Tc-99—1.36×10^{-2} µCi/L.

5.2.1.3   *Waste Volume Leaked to Soil.* There was no attempt to keep a mass balance on the vault solution. Changes in the solution levels in the vault did not show signs of gross leakage, and no reports of leakage were filed. The following is an attempt to bound the amount of leakage that might have been missed by false attribution to evaporation. Liquid on process cell floors has shown that ~1/4 in./day of liquid is lost to evaporation to the cell ventilation air. The vault floor area is 15×80 ft ($1,200$ ft^2); therefore, the daily evaporation loss could amount to 25 ft^3 (188 gal or 713 L). This is a high estimate, because the WG/WH vault circulation was poor compared to the cells. The likely evaporative losses under the old ventilation system would be ~1/4 of that observed in the process cells, 50 gal, or 189 L. A reasonable upper bound for solution loss erroneously attributed to evaporation would be ~200 gal or ~757 L.

5.2.1.4   *Summary.* Release of liquid waste at the CPP-118 site might have occurred in the 1983-1984 timeframe, when liquid collected in the WG/WH tank vaults to a depth of 3 to 4 ft within the vault and unlined duct. The unlined duct penetrates the vault wall at approximately 12 in. from the vault floor; hence, it was under water. The sources of this release and chemical constituents are not precisely known but were materials that normally would have been routed to the WG/WH tanks. While the tank vaults are lined with stainless steel, the ducts are not. They are constructed of concrete without water stops at joints with the vault walls and came into contact with liquid waste when the tank vaults flooded. The bottoms of the ducts are located at a depth (57 ft below grade) where they are believed to be in contact with fractured basalt and compacted fill. The amount of liquid released is believed to be the equivalent of 200 gal of typical PEW solution of the time. Applying the typical deep tank PEW composition for the early 1980s to the liquid released at CPP-118 results in a loss of 0.152 Ci (Cs-137, Sr-90, H-3, Tc-99, and I-129). Table 5-2 shows the individual radionuclide contribution to the total.

5.2.2   Cleanup

There have been no cleanup efforts at the CPP-118 site.

Table 5-2. Radionuclides of concern released at the CPP-118 site.

Radionuclide	Radioactivity per L (Based on Typical Process Equipment Waste Concentrations) (µCi/L)	Radioactivity Released Assuming 757 L of Waste Released (200 gal)
Cs-137	100	0.0757 Ci
Sr-90	100	0.0757 Ci
H-3	1	7.57×10^{-4} Ci
I-129	1.60×10^{-3}	1.21×10^{-6} Ci
Tc-99	1.36×10^{-2} [a]	1.03×10^{-5} Ci
	Total	0.152 Ci

a. Activity is based on the Cs-137-to-Tc-99 ratio for 5-year-old fuel using the ORIGEN2 (Croff 1980) computer simulation results for the radionuclide products and decay of a nuclear fuel.

5.2.3　Previous Investigations

There have not been any previous investigations of the CPP-118 site. The physical construction of the vent ducts was not recognized as a potential leak source when the tank vault floods occurred. The vault ventilation system has since been upgraded with new blowers and ducting, and the original ducts were sealed with stainless-steel plates in 1991.

5.2.4　Contamination Remaining at the CPP-118 Site

The 200 gal of PEW was released directly into the basalt at a depth of 53 ft bgl. Because the area is under the CPP-601 building, no removal efforts were attempted.

5.2.4.1　Areal and Vertical Extent. The areal extent of contamination is expected to be limited to the area immediately below the vent ducts, approximately 3×10 ft. The vertical extent begins at the highest level of the vault flood, 53 ft bgl, and extends vertically and laterally unknown distances into the fractured basalt.

5.2.4.2　Remaining Curies. The radionuclides of concern released totaled 0.152 Ci, as presented above. The only removal of source material has been radioactive decay. The activity is dominated by Cs-137 and Sr-90 with half-lives of 30 years and 28.8 years, respectively. Therefore, approximately 70% of the originally released material would still be present.

5.2.4.3　Uncertainties/Data Gaps. There are major uncertainties associated with this site, but the release is relatively small. The volume released is the biggest unknown followed by the contaminant activities in the liquid released. The typical PEW composition for the early 1980s provides a reasonable estimate of radionuclide activities in the released liquid. Even if the release volume was an order of magnitude higher, the curies lost would total a little over 1.5 Ci.

5.3　CPP-119, P-Cell Wall Drain

The CPP-119 release site is located in the CPP-602 building under the P-cell floor at approximately 32 ft bgl.

5.3.1　Description of Release

Condensate from a second-cycle product evaporator (P-110) located in the P-cell was discharged to a 2-in., stainless-steel drain line (2"P-1138C) that directed the waste to the WH/WG deep PEW tanks in CPP-601 (Figures 2-2, 4-1, and 5-2). Other PEW sources in P-cell (overflow to vessel off-gas prior to 1984, decontamination of cell floor or sample stations) also used this drain line on an infrequent, episodic basis.

Examination of the vertical run portion of the drain line with a borescope camera found that approximately 10 ft down the line transitioned to horizontal at a tee fitting. The weld roots were observed to be corroded and one weld approximately 1 to 2 ft above the tee appeared not to have been completed, showing evidence of three tack welds, but no circumferential welding. Solutions, primarily condensate from the evaporator, were directed down the line, a portion of which was released to the soil surrounding the unfinished weld.

5.3.1.1　Background of System Configuration and Leak. During the second-cycle process, an evaporator was used to concentrate the uranium solution for most efficient processing. The P-110 evaporator housed in the P-cell was used on the second-cycle product while the Q-110 evaporator was

used in the Q-cell to concentrate third-cycle product. The condensate generated from the operation of P-110 was piped through a 2-in., stainless-steel drain line (2"-1138C) as a PEW stream. The line was run into the south cell wall, joining a wall drain from the N-cell. At the junction, the drain traveled vertically to a depth of approximately 38 ft bgl where it was connected into a horizontal drain line with a tee. The horizontal line connected to an additional P-cell drain on one end and to the PEW CPP-601 waste trench header on the other. The N-cell drain and additional P-cell drain were rarely used and, if used, received decontamination water.

The uncompleted weld was discovered during the summer of 1989 as a result of a comprehensive evaluation of process lines. The borescope camera allowed operations personnel to determine drain system integrity on portions not visible for external inspection. Based on the condition of the weld, it was believed that a portion of the P-110 condensate and other PEW solutions leaked through the unwelded pipe joint into the soil. Because the weld was not completed, the line had been used in that condition since 1953, totaling 36 years. The discovery was made while all plant operations were suspended to allow the buried waste lines to be investigated for RCRA compliance. This drain and the rest of the buried lines were taken out of service immediately and were replaced with RCRA-compliant piping before operations were resumed in late 1991 (INEEL 2002).

During operation of the P-110 evaporator, condensate flow rates were relatively steady at 20.25 L/hr because of fairly constant operating conditions. The condensate was acidic (pH<2) and contained mercury and other heavy metals, hexone, and relatively low activities of mixed fission products. Other PEW source volumes are more difficult to estimate. If we assume two overflows to the vessel off-gas occurred in each of the 81 campaigns prior to 1984, each of ~20 L, a total of ~ 2,250 L can be estimated from that source. Decontamination efforts in P-cell were infrequent, but an approximate total of 4,000 L of decontamination solution was estimated. The decontamination solutions can be conservatively bounded as typical PEW solution.

5.3.1.2   *Waste Source Term.* The P-110 condensate liquid concentration estimates (Appendix A) were calculated through mathematical modeling based on historical process flowcharts A.

The estimated curies in the condensate are as follows:

- Cs-137—1.19×10^{-4} μCi/L

- Sr-90—8.47×10^{-5} μCi/L

- H-3—$2.55 \times 10\text{-}7$ μCi/L

- I-129—2.73×10^{-5} μCi/L

- Tc-99—2.73×10^{-4} μCi/L.

The overflow and decontamination solutions will be conservatively modeled as typical PEW solutions. Even though the volumes of these solutions were relatively small, their contribution dominates the radioactive contribution to the source term.

5.3.1.3   *Waste Volume Leaked to Soil.* The amount of waste that leaked through the faulty weld is uncertain. However, 3% of the total flow appears to be reasonable, because the pipe is vertically oriented, the system did not run pressurized, the flow would have been a thin sheet of liquid running down the pipe wall, and the liquid would experience resistance to flow out of the pipe as a result of soil-entry friction losses.

A total flow can be calculated using the number of fuel reprocessing campaigns and their average duration. Assuming that there were 42 campaigns that ran around the clock for 8 weeks, each at a flow rate of 20.25 L/hr would produce a total of 1,143,100 L (300,800 gal) of second-cycle condensate. Decontamination and overflow solutions routed to PEW accounted for another 6,250 L (1,645 gal). A 3% loss through the faulty weld translates into approximately 34,500 L (9,000 gal) over the course of 36 years.

5.3.1.4   *Summary.* The CPP-119 site was discovered during pipe inspection activities during the summer of 1989. A borescope camera investigation conducted on the P-cell wall drain determined that one of the welds on a vertical portion of the 2-in., stainless-steel drain line was incomplete. Three tack welds held the joint together, but there was no evidence that the weld was ever completed after it was tacked together. The drain line was used to route second-cycle condensate from the P-110 product condenser to the WH/WG deep PEW tanks. The drain line was used over a 36-year period in which 42 separate fuel-reprocessing campaigns were completed. During operation of the P-110 condenser, condensate flow rates were 20.25 L/hr based on documented flow sheet data. The liquid released to the soil surrounding the faulty weld was estimated to be 34,500 L (3% of the total flow), knowing the configuration of the piping and the fact that the drain was not pressurized. Individual radionuclide contribution to the total is shown in Table 5-3.

5.3.2   Cleanup

No cleanup activities were attempted because of the release location being under the building floor and foundation at 38 ft bgl. The 2"P-1138C drain line was removed from service after the faulty weld was discovered.

Table 5-3. Radionuclides of concern released at the CPP-119 site.

Radionuclide	Radioactivity per L (Based on Appendix A) (μCi/L)	Radioactivity Released Assuming 34,000 L of Waste Released (μCi)
Cs-137	1.19×10^{-4}	4.05
Sr-90	8.47×10^{-5}	2.88
H-3	2.55×10^{-7}	8.66×10^{-3}
I-129	2.73×10^{-5}	0.93
Tc-99	2.73×10^{-4} [a]	9.29
	Total	17.16 μCi

a. Activity is based on the Cs-137-to-Tc-99 ratio for 5-year-old fuel using the ORIGEN2 (Croff 1980) computer simulation results for the radionuclide products and decay of a nuclear fuel.

5.3.3   Previous Investigations

No investigations were completed on this release.

5.3.4 Contamination Remaining in Alluvium

Assuming the loss of 9,000 gal of P-110 condensate, the volume of soil impacted would be approximately 4,800 ft^3, assuming a porosity of 25% and saturated conditions. This equates to a cube of soil, 16.9 ft per side. The leak occurred at approximately 38 ft bgl, leaving 14 ft of alluvium between the point of release and the top of basalt believed to be located at 52 ft bgl. Assuming a vertical, conical-shaped plume with a 30-degree, off-vertical lateral dispersion factor, the 9,000 gal of condensate would have wetted a volume of 960 ft^3 (20% of the total possible) before the wetting front encountered the basalt. The remainder of the condensate (80%) would have entered the basalt.

5.3.4.1   *Areal Extent.* The areal extent of contamination within the alluvium at this site is limited to approximately 200 ft^2, assuming a conical-shaped distribution of contamination. Distribution of contamination within the basalt is difficult to estimate based on the fractured nature of the bedrock.

5.3.4.2   *Remaining Curies.* The amount of curies totaled approximately 17.2 μCi. The only removal of source material has been radioactive decay. A portion of contamination is from Cs-137 and Sr-90 with half-lives of 30 years and 28.8 years, respectively. Therefore, approximately 80% of the originally released material would still be present.

5.3.5 Uncertainties/Data Gaps

The biggest uncertainty associated with the CPP-119 release site is the volume lost. The 3% loss rate is a reasonable estimate, based on the leak configuration. However, a precise volume released to the soil will never be known.

5.4   CPP-120, CPP-601 West Vent Corridor and Drains

The CPP-120 site (see Figures 2-2, 4-1, and 5-2) is located under the CPP-601 West Vent Tunnel (WJ) from just north of the junction with the CPP-640 Vent Tunnel (HV) to the junction with the South Vent Tunnel (VT). The floor of the West Vent Tunnel is located at 8 ft bgl. The western edge of the release site is bounded by the western shielding wall of the CPP-601 cell row, which extends to between 25 ft bgl and 57 ft bgl (depth increases from north to south). The northern portion of the release site is bounded by the east wall of CPP-640, which extends 16 ft bgl.

5.4.1 Description of Release

The CPP-120 site had a series of potential releases from failed waste piping under the vent tunnel. Valve and piping failures in the West Vent Tunnel released contaminated liquids to the vent tunnel floor that could reach the soil via cold joints at the edges of the floor slab.

5.4.1.1   *Background of System Configuration and Leak.* The CPP-601 West Vent Tunnel was constructed with a flat concrete floor. The tunnel had three floor drains evenly spaced along its length that were connected by a directly buried 4-in. cast-iron line (4"WJ-1074T). The 4-in. cast-iron line also connected cast-iron drains from the West Sample Corridor directly above the vent tunnel and directed the liquid to the PEW collection tanks. The PEW collection system was described in Section 3.2. This line ran under the length of the West Vent Tunnel and then dropped down to enter the WG tank vault. Before entering the vault, the line converted to a 3-in., stainless-steel line at 34 ft bgl.

There was originally no process piping in the vent tunnel, but process piping was added as the plant evolved, especially after the construction of CPP-640. Many of the cells served by the West Vent Tunnel were used for first-cycle processes; thus, leaks, ventilation reversals, and liquid vented from a

1959 criticality incident all resulted in high levels of contamination. Subsequent use of corrosive decontamination chemicals resulted in failure of the cast-iron piping. Cells served by the northern part of the vent tunnel (Z, K, and J) did not cause heavy contamination of the adjacent vent tunnel; therefore, little, if any, corrosive solution entered the north floor drain. Thus, the piping failure was restricted to the middle floor drain and south to the transition to stainless steel.

The original flat floor did not effectively direct solution to the drains, so leaked solution would tend to form shallow puddles. Where these puddles contacted the wall edges, and particularly at the CPP-640 Vent Tunnel junction, cold joints and cracks could allow solution to leak to the soil below. Small amounts of contaminated liquid were released to the floor as condensates from the DOG vacuum control valve and from an E-DOG sample system. The major source of radioactive material released inside the West Vent Tunnel was episodic leakage associated with the PSV-88 valve in the line from the electrolytic process in Cell 5 to F-cell. Solution released to the vent tunnel floor produced fields of 50 to 100 R/hr. The failure of the cast-iron piping was first detected during decontamination of the vent tunnel in 1979, also marking the first confirmed release in the soil below the vent tunnel.

Remedial efforts at this time included removing a layer of contaminated concrete from the vent tunnel floor and covering the remaining hot spots with lead. Three new stainless-steel drains were installed near the sites of the old failed drains by slant drilling into the adjacent process cells (K, F, and C) and running new stainless-steel drain pipes to connect with existing PEW lines in the cells. The old drains were abandoned in place, and the lower end of the drain line was capped in the WG tank vault. A new floor was poured over the old slab and shielding. Unfortunately, the new floor was also flat and did not effectively direct liquid to the drains. The new piping did not have secondary containment; therefore, the vent tunnel still did not meet RCRA requirements for leak detection or secondary containment.

A second major contamination of the West Vent Tunnel occurred in late 1981 when PSV-88 failed during decontamination of Cell 5 after an electrolytic dissolution campaign (Zohner 1996). As before, the solution released to the vent tunnel floor produced fields of 50 to 100 R/hr. A decontamination spray system had been installed and connected to the PM area decontamination chemical header in 1979. This system aided decontamination of the 1981 releases. The new stainless-steel drains had seen little service and were not likely to have leaked. Unfortunately, the valves used to isolate the decontamination spray system leaked badly afterward, as was discovered about 3 years later during a vent tunnel entry. Examination of the floor showed extensive damage caused by acid dripping from the spray nozzles. Pits up to 3 in. deep and etched flow channels were found along with stains from liquid escaping the DOG vacuum control valve. Immediate corrective action disconnected and capped the decontamination spray system.

Later corrective action occurred in 1990–1991 during the Buried Lines Replacement Project (INEEL 2002). The existing drain lines were capped, the floors of the CPP-601 and CPP-640 Vent Tunnels were contoured to drain to new stainless-steel sumps, and the sumps were equipped with leak-detection instruments and air jets to remove liquid. A new PEW collection header was installed in the vent tunnel. This line collected the discharge of the sump jets, the sample corridor drains, and PEW solutions from CPP-640 and transferred them to the PEW tanks via RCRA-compliant piping.

5.4.1.2 *Waste Source Term.* The major leakage in the West Vent Tunnel originated with the failure of the PSV-88 valve in the dissolver product transfer line from Cell 5 to F-cell. The radionuclide content of the flush solutions used at the start of Cell 5 cleanup in late 1981 can be estimated from data on the decontamination of G- and H-cell in 1982 (EDF-5318). It is expected that both the size and composition of the 1970s leakage would be similar. There are no direct data on DOG condensate, but the stains left by it were not highly radioactive; thus, this solution can be conservatively modeled by combining it with the decontamination flush solutions.

There are no analytical data for the leaked liquids at the CPP-120 site. However, the activity can be approximated from historical sources. The activity levels for the radionuclides of concern in the lost dissolver product solution are based on dissolver product derived from fuels with typical fission yields:

- Cs-137—3.5 Ci/L

- Sr-90—3.5 Ci/L

- H-3—1.41×10^{-2} Ci/L

- I-129—7.91×10^{-7} Ci/L

- Tc-99—4.76×10^{-4} Ci/L.

The activity level for the leaked decontamination flush solution is based on 1% concentration of the original dissolver product. The arbitrary 1% solution value was selected based on the fact that the process equipment was designed to be fully drained, leaving minimal heels in the vessels and piping. It was reasoned that, volumetrically, 1% of the process liquid could reside in the drained equipment that would combine with the decontamination fluid once it was introduced into the system. Therefore, the decontamination liquid would contain 1/100th of the original dissolver product activity:

- Cs-137—0.035 Ci/L

- Sr-90—0.035 Ci/L

- H-3—1.41×10^{-4} Ci/L

- I-129—7.9×10^{-9} Ci/L

- Tc-99—4.76×10^{-6} Ci/L.

5.4.1.3 *Waste Volume Leaked to Soil.* Process leaks from the electrolytic dissolver transfer line in the vent tunnel were small enough not to trigger concerns for uranium loss; therefore, this leakage source did not exceed a few liters. Leakage from this source would have been readily detectable in routine PEW samples before the drains failed; thus, 10 L is a reasonable bound for these leaks. The first major leakage occurred at nonspecific times after the 1971 installation of the electrolytic process through 1977, with evidence that more than one event of this nature had occurred. A similar leak occurred during the start of chemical decontamination procedures for Cell 5 in CPP-640 in late 1981. Then, volume discrepancies on transfers between Cell 5 and F-cell led to the discovery of the leakage in the vent tunnel. A reasonable bounding value for the decontamination solution leakage would be 25 gal (~100 L).

Condensate leaks from the DOG system between the drain line replacement and the 1990 upgrades for RCRA compliance led to pitting and staining of the floor but had small volume (~4 L based on the size of the stains). This indicates that the earlier leakage from these sources also was small, contributing to the degradation of the drain piping but only amounting to a few liters. We will assume that 20 L is a conservative bound for the DOG condensate. Thus, the source leakage to the vent tunnel floor can be modeled as 10 L of dissolver product, 100 L of decontamination flush solution from Cell 5, and 20 L of DOG condensate.

This radioactive source material was then flushed with several hundred liters of water and nitric acid, which could have helped carry the source material into the soil due to the failed piping. Personnel entering the vent tunnel for cleanup work delivered flush solutions via garden hose or carboy. A portion of the flush solution was recovered and transferred to the deep tanks. No volume records were kept, but

300 gal (1,140 L) are assumed as a conservative bound for the volume of flush solution, which might have leaked to the soil.

This estimate yielded 130 L of radioactive liquid, followed by 1,140 L of nonradioactive flush solution (water with some nitric acid). These numbers were associated both with the leak event in the West Vent Tunnel during the 1970s and again with the leak in 1981. Only leaks prior to 1979 had direct access to the failed cast-iron piping. Leaks after 1979 were confined to the vent tunnel floor, and a relatively small portion of the solution likely would have reached the ground through cold joints. These incidents are minor relative to the volume and uncertainty associated with the primary release.

5.4.1.4   Summary. Solution leaks and subsequent decontamination efforts caused the failure of buried cast-iron waste piping under the CPP-601 West Vent Tunnel. The major solution releases to the soil occurred between 1971 and 1978 during electrolytic campaigns and Cell 5 decontaminations. Solution released at the CPP-120 site was a mixture of electrolytic dissolver product, hot decontamination solutions from cleanup of Cell 5 in CPP-640, and cold decontamination solutions used in the vent tunnel. Some solutions were released to the vent tunnel floor at other times and might have reached the soil through cold joints at the edges of the floor slab, but these releases would have been small compared to those involved in the failure of the cast-iron drains. The total radioactivity of the liquids amounts to 148.0 Ci for the five radionuclides of concern and are summarized in Tables 5-4 and 5-5.

5.4.2   Cleanup

The vent tunnel was subject to extensive decontamination, but there have been no cleanup efforts under the vent tunnel floor at the CPP-120 site.

Table 5-4. Radionuclides of concern released at the CPP-120 site from the dissolver product.

Radionuclide	Radioactivity per L (Based on Dissolver Product Derived from Fuels with Typical Fission Yields) (Ci/L)	Radioactivity Released Assuming 10 L of Waste Released (5.3 gal)
Cs-137	3.5	35 Ci
Sr-90	3.5	35 Ci
H-3	1.41×10^{-2} [a]	0.141 Ci
I-129	7.91×10^{-7} [a]	0.79×10^{-5} Ci
Tc-99	4.76×10^{-4} [a]	4.76×10^{-3} Ci
	Total	70.15 Ci

a. Activity is based on the Cs-137-to-I-129, Cs-137-to-H-3, or Cs-137-to-Tc-99 ratio for 5-year-old fuel using the ORIGEN2 (Croff 1980) computer simulation results for the radionuclide products and decay of a nuclear fuel.

Table 5-5. Radionuclides of concern released at the CPP-120 site from the decontamination liquid.

Radionuclide	Radioactivity per L (Based on Average Zircon-Clad Fuel Dissolver Product) (Ci/L)	Radioactivity Released Assuming 120 L of Waste Released (31.6 gal)
Cs-137	0.035	4.2 Ci
Sr-90	0.035	4.2 Ci
H-3	1.41×10^{-4} [a]	1.69×10^{-2} Ci
I-129	7.91×10^{-9} [a]	9.49×10^{-7} Ci
Tc-99	4.76×10^{-6} [a]	5.71×10^{-4} Ci
	Total	8.40 Ci

a. Activity is based on the Cs-137-to-I-129, Cs-137-to-H-3, or Cs-137-to-Tc-99 ratio for 5-year-old fuel using the ORIGEN2 (Croff 1980) computer simulation results for the radionuclide products and decay of a nuclear fuel.

5.4.3 Previous Investigations

The buried PEW lines under the West Vent Tunnel were investigated for RCRA compliance as part of the Buried Lines Replacement Project between 1989 and 1991. The abandoned cast-iron lines were not disturbed, since their lack of integrity prohibited flushing. The replacement stainless-steel drains to the cells were not RCRA-compliant, so they were capped off and abandoned; their short period of use made it likely that they had never leaked. The new vent tunnel floors, sumps, and PEW collection piping were installed in 1991.

No soil samples were taken, because the original response to the pipe failure was to assume that the lines had simply plugged. Later experience indicated that horizontal cast-iron pipe tends to fail by collapsing, effectively blocking flow. In addition, during the 1991 work, contamination was found to be entering the CPP-640 basement from the area of the CPP-120 site release. Moisture entering at the crack between the Cell 5 wall and the CPP-601 West Vent Tunnel footing evaporated leaving whitish deposits with 40,000 to 50,000 dps/100 cm² contamination.

5.4.4 Contamination Remaining at the CPP-120 Site

5.4.4.1 Areal and Vertical Extent. The areal extent of contamination is expected to be limited to the area immediately below the West Vent Tunnel from the south wall of G-cell south to the WG vault, approximately 130×10 ft. The vertical extent begins at the level of the vent tunnel floor, 8 ft bgl, and extends downward into the compacted fill. Assuming that 300 gal (1,140 L) of water escaped confinement while flushing the contamination from the West Vent Tunnel, along with the 34 gal (130 L) of contaminated liquid released, the total volume controlling the movement of contamination is 334 gal (1,264 L). Because the release occurred 8 ft bgl, the leak has most likely been contained within the 45 ft of alluvium below.

The volume of soil that would contain the 334 gal of liquid at a porosity of 25% would be 178 ft^3. Assuming a line source that is 130 ft long, the liquid would migrate to a depth between 1.5 and 2.0 ft. If the release were a point source, the conical-shaped plume would reach a depth of approximately 8 ft assuming a lateral spread of 30 degrees off vertical. Therefore, the release appears to be contained within the alluvium.

5.4.4.2   Remaining Curies. The only removal of source material has been radioactive decay. The activity is dominated by Cs-137 and Sr-90 with half-lives of 30 and 28.8 years, respectively; thus, approximately 70% of the 147 Ci would still be present.

5.4.5   Uncertainties/Data Gaps

The constituents in the released liquids are fairly well understood and the activity levels for the two waste streams are reasonable. The volume of contaminated liquid lost also appears to be reasonable based on the fact that system uranium loss was monitored closely. Uranium loss would be indicative of a measurable release, and no losses were indicated. Therefore, it is believed the release of highly contaminated fluid was relatively small.

5.5   CPP-121, M-Cell Floor Liner

The CPP-121 site is located under M-cell of the CPP-601 building (see Figures 2-2, 5-1, and 5-2). The M-cell is located near the south end of the east cell row. The M-cell floor is a 1-ft-thick concrete slab poured over compacted fill, whose top elevation is 32 ft bgl.

5.5.1   Description of Release

A small hole in the M-cell floor liner approximately 3 in. above the floor was identified when the floor was flooded with water during performance of criticality safety tests prior to the first use of M-cell in 1983 (Zohner 1996). While there are no events identified that would have subsequently flooded M-cell to this level, it is possible that during cell decontamination activities, small amounts of radioactive materials could be forced into this hole.

5.5.1.1   Background of System Configuration and Leak. The original construction of the M-cell floor was the same as that used for all process cells in CPP-601. First, the wall footings and shielding walls were poured. Then, the floor area fill was leveled and compacted, and the 1-ft-thick flat floor slab was poured, with cold joints where the slab overlapped the footings. The floor was then contoured with grout, up to 6 in. thick at the high points, to produce a surface that drained smoothly to a low area at the floor sump and drain. The sump was a 15-in. section of 5-in. pipe. The floor grout and the lower walls were poured with embedded stainless-steel studs and angle iron to allow installation of the floor liner. The stainless-steel liner plates were attached to the embedded metal with plug or seam welds.

When water was added to the cell floor as verification that sufficient borated Raschig rings were on the cell floor to contain a spill of specified size, water slowly leaked into the cell labyrinth from the underside of the cell liner. (The Raschig rings were small-diameter rings similar in shape to napkin rings that were used to control criticality of any liquids leaked onto the cell floor.) The leak did not appear until at least 3 in. of water had been added to the cell floor. This slow leak could not be readily located, and further investigation would have required lifting the floor gratings and digging in the glass Raschig rings. It was decided that the effort was not justified.

The M-cell contained two pairs of tanks for uranium accountability measurements: VES-M-103 and VES-M-104 measured solutions transferred from first-cycle extraction to intercycle storage; VES-M-101 and VES-M-102 measured solutions transferred between third-cycle and final product storage. Each tank had an associated pump. In March 1984, the M-202 pump developed a leak. The pulse action of the diaphragm pump sprayed the wall with third-cycle uranium solution, which ran down into the Raschig rings and filled the cell sump. Solution loss was limited to 1 or 2 gal, which was returned to the process via the sump process jet to VES-M-103. The cell was entered, and the wall and Raschig rings were rinsed with a water spray to complete the uranium recovery. This more-dilute uranium solution was

routed to the uranium salvage system via the sump recycle jet. The flush solutions were not allowed to accumulate on the floor, because that would have spread the contamination. The only possibility of solution reaching the pinhole in the liner would have been if the hole had been subject to direct spraying or if the hole was in the path of the solution running down the wall. Neither mechanism was likely to cause a significant amount of release behind the liner.

Another known solution release in M-cell was associated with a release of about 100 gal of cold aluminum nitrate-test solution and 100 gal of cold test water to the cell floor shortly after the flood test described above. This solution did not reach the level of the hole in the liner.

There were several small solution releases associated with repairs and replacement of the M-203 and M-204 pumps. These were limited to 1 or 2 L each, and the solutions were flushed to the sump and not allowed to accumulate.

5.5.1.2   Waste Source Term. There are no direct analytical data for the solution that might have leaked from the M-cell floor at the CPP-121 site. The only liquid known to have entered the subliner space was clean water. The only process solution that had a chance to reach the liner leak was third-cycle product solution that leaked from the M-202 pump. Third-cycle product consisted of 0.13 M nitric acid, high uranium (350 to 420 g U/L), with low levels of contamination from transuranics (5×10^{-4} mCi/L Pu), and fission products (2×10^{-4} mCi/L Cs-137 plus Sr-90) (EDF-5441). Because the process leak was small, ~2 gal or 10 L, the total amount of nonuranium activity released to the floor was much less than 1 mCi. The uranium released amounted to ~4 kg. Jetting from the sump returned nearly all uranium products to the process. Assuming that a very conservative 10% of the release got behind the cell liner, this would place 1×10^{-6} Ci total nonuranium activity and 400 g of uranium behind the liner and available to leak to the soil from cold joints around the edges of the floor slab.

5.5.1.3   Waste Volume Leaked to the Soil. The volume of solution that entered the subliner space was not known with certainty. When water was detected leaking into the cell entry labyrinth, the floor-flooding test was repeated at least twice. The volume of the subliner space is $\{[(16 \times 15 \times 1/48 \text{ ft}) - (16 \times 3 \times 1/48 \text{ ft})] \times 7.5\}$ or 30 gal. There was always a significant time delay for the appearance of water in the labyrinth, indicating that the subliner space had drained between tests. The level of the water in the cell dropped about ½ in. before stabilizing. Therefore, we could estimate the 60 gal of water leaked to the soil during each test, say 250 gal total. This volume was clean water before process solutions were introduced to the cell. The volume of process solution that entered the subliner space was conservatively estimated at 0.25 gal.

5.5.1.4   Summary. Clean water leaked behind the M-cell liner during preprocess testing. A small release of process solution might have allowed ¼ gal of radioactive solution to leak behind the liner. The behavior of the leaking water indicated that the subliner space did release solution to the soil. The release can be bounded at 250 gal of clean water and 0.25 gal of process solution. No liquid volume has been allowed to accumulate on the M-cell floor sufficient to reach the hole in the liner since the original tests.

5.5.2   Cleanup

There have been no cleanup efforts at the CPP-121 site.

5.5.3   Previous Investigations

There have been no previous investigations at the CPP-121 site.

5.5.4   Contamination Remaining at the CPP-121 Site

5.5.4.1   Areal and Vertical Extent. The areal extent of the contamination is expected to be limited to the area directly below M-cell, approximately 15 × 16 ft. The vertical extent of contamination begins in compacted fill beneath the cell floor, 32 ft bgl, and continues down to the basalt surface at approximately 52 ft bgl.

5.5.4.2   Remaining Curies. The only removal of source material has been radioactive decay. The activity is dominated by Cs-137 and Sr-90 with half-lives of 28 years and 30 years, respectively; thus, approximately 70% of the originally released material would still be present. Uranium and plutonium would not be reduced by decay; therefore, all that was released would still be present.

5.5.5   Uncertainties/Data Gaps

There is significant uncertainty whether any process contaminants reached the soil at the CPP-121 site. Estimates given are very conservative, especially for uranium.

5.6   CPP-122 Site, E-Cell Liner Pin Holes

The CPP-122 site is located under E-cell of the CPP-601 building. The E-cell is located in the west cell row, a little south of the middle of the building (see Figures 2-2, 5-1, and 5-2). The E-cell floor is a 1-ft-thick concrete slab poured over compacted fill, whose top elevation is 22 ft bgl.

5.6.1   Description of Release

During the 1980–1981 zirconium and coprocessing fuel dissolution campaigns, a release of dissolver product created pinhole leaks in the floor liner. Attempts were made in May 1986 to decontaminate the liner using heated nitric acid on the cell floor (depth of 6 to 12 in. for a period of 2 weeks). This effort would have also released these materials to the concrete under the liner, where they could reach the soil via cold joints at the edges of the slab.

5.6.1.1   Background of System Configuration and Leak. The E-cell contained VES-E-101, the zirconium fuel dissolver. Zirconium-clad fuel was dropped into the dissolver from the charging chute located in the PM area. A hydrofluoric-acid-based dissolvent solution was made up in the PM area and metered into the dissolver from a PM area feed tank. Both the tanks and feed lines required special construction to withstand the highly corrosive hydrofluoric acid. The tanks were lined with rubber, and much of the feed system was constructed of KEL-F, a chemically resistant plastic. The two lines entering the cell and VES-E-101 were made of Monel, an alloy resistant to hydrofluoric acid, but not to nitric acid. The E-101 vessel had been replaced shortly before the start of the campaign, and short lengths of piping connecting the new dissolver to process and utility lines also were new. Apparently, an 18-in. section of the primary hydrofluoric acid dissolvent feed line was installed with the wrong material—ordinary stainless-steel instead of Monel. When hydrofluoric acid feed was started to the dissolver, this material corroded away within a short time.

Material was released in three stages. First, cold chemical dissolvent (acidic [pH substantially <0], high fluoride [4 M, corrosive to stainless steel], nonradioactive) being routed to the E-101 dissolver leaked through the hole in the transfer pipe and ran across the floor to the cell sump. This material was detected in the cell sump, and the cause was determined by observing that the cell sump began rising soon after a dissolvent transfer began. The amount of dissolvent lost was limited to about 5 gal, because the dissolvent transfers were shifted to the alternate line as soon as the leak source was identified.

The second stage of the leak consisted of very highly radioactive first-cycle dissolver product that splashed out of the leaking pipe during dissolver operation. This solution was similar to the cold stream dissolvent in acidity and fluoride content, but now contained highly radioactive fission products (~12.2 Ci/L fission products). This leak was slow and intermittent, tending to crystallize on the floor, and only occasionally requiring sump liquid to be jetted back to the process. Radiation fields at the floor level of E-cell were in excess of 200 R/hr after the process vessels had been flushed for uranium recovery and emptied following internal decontamination.

The third stage of the leak occurred during decontamination of the cell floor in preparation for maintenance activities in E-cell during May 1986. At that time, a dam was constructed in the cell entry labyrinth, and the cell floor was flooded with nitric acid (2–4 M) to a depth of 6 to 12 in. A temporary steam jet was used to heat and recirculate the acid over a period of at least 2 weeks. Acid was removed and replenished several times during this decontamination effort. During this time, the acid might have carried some portion of the released material left on or under the floor liner into the soil below.

The original construction of the E-cell floor was the same as that used for all process cells in CPP-601. First, the wall footings and shielding walls were poured. Then, the floor area fill was leveled and compacted, and the 1-ft-thick flat floor slab was poured, with cold joints where the slab overlapped the footings. The floor was then contoured with grout, up to 6 in. thick at the high points, to produce a surface that drained smoothly to a low area at the floor sump and drain. The sump was a 15-in. section of 5-in. pipe. The floor grout and the lower walls were poured with embedded stainless-steel studs and angle iron to allow installation of the floor liner. The stainless-steel liner plates were attached to the embedded metal with plug or seam welds. The E-cell floor was modified after original construction to incorporate the footing for VES-E-101. A square hole, ~3 ft on a side, was cut through the existing floor and steel pilings were driven to bedrock. A concrete footing block was poured that extended 6 in. above the existing floor level. The block was then covered with a stainless-steel liner that was welded to the existing floor liner. No water stops were installed at the junction of the new block and the old floor concrete.

After the floor had been decontaminated with nitric acid, the decontamination workers who entered the cell noticed that liquid would squirt up through pinholes in the cell liner when they stepped in some places. These holes were produced by the corrosive action of the released hydrofluoric-acid-based solutions on stainless steel. The areas of damaged liner were relatively small, but the observed effects indicate that the space under the liner was filled with liquid. The volume under the liner (20 ft × 19 ft × ¼ in.) amounts to a maximum of 8 ft^3 or 60 gal. The volume of acid added to flood the floor to a 1-ft depth over the sump would be 19 × 20 × 0.5 ft [top 6 in.] + (19 × 20 × 0.5 ft)/2 [sloped bottom 6 in.] = 285 ft^3. There are 7.5 gal/ft^3 (285 ft^3 × 7.5 gal/ft^3 = 2,137.5 gal) and 60 gal in the labyrinth; thus, the total is approximately 2,200 gal.

5.6.1.2   *Waste Source Term.* There are no analytical data for the dissolver product that might have leaked from the E-cell floor at the CPP-122 site. However, the activity in the waste can be estimated from the historical analyses of the solutions known to have been used in E-cell (EDF-5441). The activities in the dissolver product for the five radionuclides of concern are as follows:

- Cs-137—3.5 Ci/L

- Sr-90—3.5 Ci/L

- H-3—1.41 × 10^{-2} Ci/L

- I-129—7.91×10^{-7} Ci/L

- Tc-99—4.76×10^{-4} Ci/L.

5.6.1.3   *Waste Volume Leaked to the Soil.* Because there was no attempt to keep a mass balance on the decontamination solution, the volume lost to the soil is not known with certainty. However, the total dissolver product solution leaked to the floor can be bounded at approximately 30 gal (114 L), based on jetting the 1-gal sump every few days. It is likely that most of this material dried on the cell floor, and this amount of radioactive waste could account for the observed 200-R/h fields.

Changes in the decontamination solution levels on the cell floor did not show signs of gross leakage, but two effects would have made volume changes hard to interpret. The steam jet would add condensate to the liquid on the floor and raise its temperature. The warm liquid also would evaporate. Observation of liquid on process cell floors has shown that ~1/4 in./day of liquid is lost to evaporation to the cell ventilation air. The E-cell floor area is 19×20 ft (380 ft^2); thus, the daily evaporation loss could amount to 8 ft^3 (60 gal, 228 L) or 2.5 gal/h. This amounts to just under 3% of the liquid volume on the floor. The steam jet used to heat and circulate the liquid could add 25 gal/h of steam condensate during the time the jet was run. This addition could mask the loss of approximately 1% of the solution per hour or 8% in a typical shift. This does not prove whether solution was lost to leakage but serves to illustrate the error bounds that were in existence. Based on these facts, a 10% leakage is a reasonably conservative estimation for potential solution loss. This uncertainty amounts to 3 gal of dissolver product and 220 gal per flush or about 1,000 gal for the decontamination effort.

5.6.1.4   *Summary.* Contaminated solutions leaked to the floor of E-cell during operations in 1980 and 1981. Later decontamination efforts in 1986 mobilized the radioactive waste and might have allowed some to leak to the soil below the cell. Reasonable estimates for the leakage place the potential release at 3 gal of dissolver product contained in up to 1,000 gal of 4 M nitric acid. The activity released to the soil is summarized in Table 5-6.

5.6.2   Cleanup

There have been no cleanup efforts at the CPP-122 site.

Table 5-6. Radionuclides of concern released at the CPP-122 site from the dissolver product.

Radionuclide	Radioactivity per L (Based on Average Zircon-Clad Fuel Dissolver Product) (Ci/L)	Radioactivity Released Assuming 11 L of Waste Released (3 gal)
Cs-137	3.5	38.5 Ci
Sr-90	3.5	38.5 Ci
H-3	1.41×10^{-2} [a]	0.16 Ci
I-129	7.91×10^{-7} [a]	8.70×10^{-6} Ci
Tc-99	4.76×10^{-4} [a]	5.24×10^{-3} Ci
	Total	77.16 Ci

a. Activity is based on the Cs-137-to-I-129, Cs-137-to-H-3, or Cs-137-to-Tc-99 ratio for 5-year-old fuel using the ORIGEN2 (Croff 1980) computer simulation results for the radionuclide products and decay of a nuclear fuel.

5.6.3 Previous Investigations

There have been no previous investigations at the CPP-122 site, because the significance of damage to the cell liner was not recognized at the time of decontamination; therefore, it was believed that all liquid was confined within E-cell.

5.6.4 Contamination Remaining at the CPP-122 Site

5.6.4.1 Areal and Vertical Extent. The areal extent of the contamination is expected to be limited to the area directly below E-cell, approximately 19×20 ft. The vertical extent of contamination begins in compacted fill beneath the cell floor at 23 ft bgl. Assuming that a total of 1,003 gal of liquid leaked into the soil, approximately 550 ft^3 of alluvial material with a porosity of 25% would be necessary to contain the liquid. If the release occurred at a single point, the conical plume would migrate downward about 11.5 ft or to a depth of 34.5 ft bgl. However, the release likely occurred at multiple points under the footprint of the cell with the liquid residing in the first few feet below the cell floor.

5.6.4.2 Remaining Curies. The only removal of source material has been radioactive decay. The activity is dominated by Cs-137 and Sr-90 with half-lives of 30 and 28.8 years respectively; thus, approximately 70% of the originally released material would still be present.

5.6.5 Uncertainties/Data Gaps

The biggest data gap for this release site is the volume of liquid released to the subsurface. The release of dissolver product involved a small volume within the cell, most of which was recovered. The decontamination solution volumes released, on the other hand, are harder to bound.

5.7 CPP-123, F-Cell Sump

The CPP-123 site is located under F-cell of the CPP-601 building. The F-cell is located in the middle of the west cell row (see Figures 2-2, 5-1, and 5-2). The F-cell floor is a 1-ft-thick concrete slab poured over compacted fill, whose top elevation is 22 ft bgl.

5.7.1 Description of Release

During the 1980–1981 zirconium and coprocessing fuel dissolution campaigns, a release of highly contaminated E-DOG scrubber solution occurred from the E-DOG scrubber pumps located in F-cell. The leak caused very high beta radiation fields in F-cell that were controlled by flooding the cell floor with water. Some of this water leaked under the stainless-steel floor liner through an unsealed hole for a bolt that secured the cell ladder to the floor, carrying some of the highly radioactive waste with it. The liquid under the liner resulted in a 5-R/h gamma field in the lower portions of the cell near the sump. Some of this liquid might have been released to the soil at cold joints around the edges of the floor slab. An attempt to flush the subliner space during 1991 might have forced additional waste into the soil.

5.7.1.1 Background of System Configuration and Leak. The source liquid originated in VES-E-104, whose function was to remove hydrofluoric acid vapor from the VES-E-101 DOG stream. A solution of aluminum nitrate and boric acid was sprayed onto the plate column section of VES-E-104, where the solution ran down countercurrent to the rising E-DOG stream. The solution collected in the lower section of VES-E-104 and was pumped back to the spray nozzles by pumps located on the floor level of F-cell near the cell entry.

The pumps leaked during the course of the 1980 and 1981 fuel dissolution campaigns during which VES-E-101 was used to dissolve zirconium-clad fuels. The scrubber solution collected and complexed hydrofluoric acid vapor, preventing corrosion in downstream equipment. It also collected nonvolatile fission products from droplets of mist and volatile fission products, especially Ru-106, which is an intense beta emitter. No sample data are available for this solution, but it would likely have about 10% (based on process knowledge) of the activity of the dissolver product. The leak rate was slow, allowing solution to dry on the pumps, walls, and floor before reaching the cell sump, but it was highly radioactive, especially in beta activity. Prior to attempting repair of the pumps in 1982, decontamination workers sprayed the spill area with water and nitric acid to remove loose material. The resulting solution was allowed to remain on the cell floor, and more water was added to act as shielding against the high beta radiation from the stains under the pumps.

The original construction of the F-cell floor was the same as that used for all process cells in CPP-601. First, the wall footings and shielding walls were poured. Then, the floor area fill was leveled and compacted, and the 1-ft-thick flat floor slab was poured, with cold joints where the slab overlapped the footings. The floor was then contoured with grout, up to 6 in. thick at the high points, to produce a surface that drained smoothly to a low area at the floor sump and drain. The sump was a 15-in. section of 5-in. pipe. The floor grout and the lower walls were poured with embedded stainless-steel studs and angle iron to allow installation of the floor liner. The stainless-steel liner plates were attached to the embedded metal with plug or seam welds.

Unknown at the time, there was an open hole in the floor liner at one of the bolts used to secure the floor ladder. This hole allowed some fraction of the liquid, carrying some of the radioactive waste, to run under the floor liner. The volume was small enough to be contained in an area about 5 ft square around the cell sump. After the pumps were removed and the lines capped, the solution was jetted from the F-cell floor to PEW. The area was decontaminated, but no amount of effort was able to reduce the 5-R/h gamma field around the sump area.

The area remained unchanged until the Buried Line Replacement Project installed a new PEW line during 1991 (INEEL 2002). While a core drill for the new line was being bored through the cell wall at floor level near the sump, water from the core-drilling machine began to appear on top of the F-cell liner. This initial water carried radiation readings of 14 R/h. Investigators were unable to determine where the water was penetrating the liner. A blue dye was added to the subliner space through a drilled hole within a 1-in. pipe stub welded to the liner. A garden hose was attached to the pipe to allow the dye to be flushed through the penetration, which was discovered at an unsealed anchor bolt for the cell ladder. The initial liquid was over 10 R/h. With continued flushing, the liquid dropped to <100 mR/h. After this flushing, dramatic reductions in the radiation fields (5,000 mR/h to 300 mR/h) around the sump were reported.

5.7.1.2   Waste Source Term. There are no analytical data for the solution that might have leaked into the soil from the F-cell floor at the CPP-123 site. However, the activity in the waste can be closely approximated by using 10% of the values found in dissolver product. This is based on the fact that the off-gas scrubber was used to remove dissolver product mist and droplets from the waste stream. The aluminum nitrate and boric acid used became contaminated over time, as it was recycled before the solution was replaced with fresh uncontaminated solution. A 10% dissolver product to 90% scrubber solution was an average mixture and would represent what was lost through the leaking pumps. The activity for the five radionuclides of concern in the E-DOG scrubber solution is assumed to be follows:

- Cs-137—0.35 Ci/L

- Sr-90—0.35 Ci/L

- H-3—1.41×10^{-3} Ci/L

- I-129—7.91×10^{-8} Ci/L

- Tc-99—4.76×10^{-5} Ci/L.

5.7.1.3   *Waste Volume Leaked to the Soil.* It is difficult to estimate the volume of solution from a slow, long-term leak under conditions that allow the solution to dry before reaching a measuring point. The residual activity remaining under the F-cell liner has been modeled, by ORIGEN2, as 4.7 Ci of Cs-137. Based on the reduction of radiation fields caused by the 1991 flush, this amount represents about 6% of the original contamination, which would mean that the original contamination would be around 80 Ci of Cs-137. This in turn would be only a small (say 8%) fraction of the original floor contamination, placing the original release at around 1,000 Ci of Cs-137. At dissolver product Cs-137 levels, this would require approximately 300 L to be released, which is possible at about 1 mL/min over a 6-month period. However, because the source liquid would be expected to only have around 10% of the activity of dissolver product, the leak rate would need to be about 10 mL/min, which is relatively high to be considered likely.

The volume of solution needed to fill the space under the F-cell liner is similar to that for the E-cell liner calculated previously. The estimate for E-cell was 60 gal; the interior shielding wall reduces this volume to 53 gal for F-cell. The volume of water added to the floor in 1981 was approximately 1,250 gal (19 × 20 × 0.88 ft [shield wall displacement] × 0.5 [accounts for slope] = 167.2 ft^3 and 167.2 ft^3 × 7.5 gal/ft^3 = 1,254 gal). If 1,000 Ci of activity were dissolved in 1,250 gal of water, 53 gal (capacity under the F-cell liner) would contain approximately 43 Ci of Cs-137, only about ½ the activity calculated above from the model. The discrepancy can be resolved by assuming either (1) that the model value is too high (some activity attributed to the subliner space might originate from sources above the liner) or (2) that a substantial transfer of the liquid to the soil did occur (bringing more activity with it to deposit under the liner). Conservatively, we assume the second. Under the assumption that 1,250 gal of water was added, bringing the liquid level up to the level of the entry labyrinth, and that the hole was about 2 in. lower, ~220 gal of liquid could drain through the hole until the liquid level dropped to the level of the hole. Assuming, from above, that the liquid contained 0.8 Ci of Cs-137/gal, this would allow 176 Ci of Cs-137 to enter the sub-labyrinth space. Because an estimated 80 Ci of Cs-137 remained under the liner, 96 Ci of Cs-137 was free to enter the soil. The calculation so far has been based on cesium gamma radiation; this amount must be slightly more than doubled to include the contribution from strontium, bringing the total potential release to 200 Ci. Based on the assumed concentration and the potential release, approximately 285 L of waste could have been released to the soil.

The volume added in 1991 did not supply any further activity but might add to the volume of the release. Because there was no attempt to keep a mass balance on either the shielding water or the flush water, the volume lost to the soil is not known with certainty. The volume of shielding water available to enter the subliner space was estimated above as 220 gal. It is a reasonable estimate that the flush water in 1991 was supplied by the garden hose at 1 gpm for a 90-min period and, again, all leaked to the soil. This would total 310 gal of liquid to the soil.

5.7.1.4   *Summary.* Contaminated solution leaked to the floor of F-cell during operations in 1980 and 1981. Later, efforts were made to shield workers by flooding the floor with water. The water carried some of the contaminated solution under the floor liner, creating a high radiation field around the sump at the low point of the floor. Some of this material might have leaked to the soil at the cold joints around the edges of the floor slab.

Later efforts to flush the contamination from under the liner might have forced additional solution into the soil by the same route. Conservative estimates place the volume of the leak at 310 gal and the radionuclides at 200 Ci (Cs-137 plus Sr-90). The activity released to the soil is summarized in Table 5-7.

Table 5-7. Radionuclides of concern released at the CPP-123 site from the E-DOG off-gas scrubber.

Radionuclide	Radioactivity per L (Based on 10% of Average Zircon-Clad Fuel Dissolver Product) (Ci/L)	Radioactivity Released Assuming 285 L of Waste Released to the Soil (75 gal)
Cs-137	0.35	100 Ci
Sr-90	0.35	100 Ci
H-3	1.41×10^{-3} [a]	0.4 Ci
I-129	7.91×10^{-8} [a]	<0.0001 Ci
Tc-99	4.76×10^{-5} [a]	0.01 Ci
	Total	200 Ci

a. Activity is based on the Cs-137-to-I-129, Cs-137-to-H-3, or Cs-137-to-Tc-99 ratio for 5-year-old fuel using the ORIGEN2 (Croff 1980) computer simulation results for the radionuclide products and decay of a nuclear fuel.

5.7.2 Cleanup

There have been no cleanup efforts at the CPP-123 site under the concrete slab. Efforts to clean the area under the floor liner in 1991 succeeded in reducing the radiation fields in the hot area near the sump by a factor of nearly 17. The contamination flushed back into the cell was transferred to PEW. However, this cleanup effort might have flushed additional waste into the soil.

5.7.3 Previous Investigations

There have been no investigations at the CPP-123 site.

5.7.4 Contamination Remaining at the CPP-123 Site

5.7.4.1   Areal and Vertical Extent. The areal extent of the contamination is expected to be limited to the area directly below F-cell, approximately 19 × 20 ft. The vertical extent of contamination begins in compacted fill beneath the cell floor, 23 ft bgl, and continues down to the basalt surface at approximately 52 ft bgl.

5.7.4.2   Remaining Curies. The only removal of source material has been radioactive decay. The activity is dominated by Cs-137 and Sr-90 with half-lives of 28 years and 30 years, respectively; thus, approximately 70% of the originally released material would still be present.

5.7.5 Uncertainties/Data Gaps

There are major uncertainties associated with this site, but the volume released is relatively small.

6. SUMMARY

The source terms for the eight past releases that have occurred within and under the FRC have been estimated in this report and are summarized in Table 6-1. The largest source volume was from the CPP-80 release at almost 400,000 L. This accounts for 91% of the total volume (440,000 L) from the eight known releases within and under the FRC. The highest activity source was from the CPP-123 release at 200 Ci. This accounts for 57% of the total curies (349 Ci) from the eight known releases within and under the FRC.

The source terms for these releases will be used to support the goal to initiate the DD&D of the FRC using the CERCLA process. The risk assessment for the non-time-critical removal action and engineering evaluation/cost analysis can be based on the conservative estimates contained in this report.

This report summarized the radionuclides of concern for the tank farm but did not present details for other radionuclides, inorganics, and organics. Although these were most likely present in the waste stream, they are expected to pose little or no risk.

Table 6-1. Radionuclides of concern released at the eight sites.

Release Site	Waste Source Term	Volume Released	Activity of Contaminants (μCi/L)					Curies Released					
			Cs-137	Sr-90	H-3	I-129	Tc-99	Cs-137	Sr-90	H-3	I-129	Tc-99	Total
CPP-80 CPP-601 vessel system off-gas condensate	VT-300 off-gas condensate	399,460 L (105,540 gal)	1.33×10^{-1}	1.33×10^{-1}	1.33	1.33×10^{-4}	1.33×10^{-3}	5.29×10^{-2}	5.29×10^{-2}	5.29×10^{-1}	5.29×10^{-5}	5.29×10^{-4}	0.635
CPP-117 Lab waste leakage through CPP-602 drain system	CPP-602 laboratory PEW waste	5,170 L (1,366 gal)	100	100	1	1.60×10^{-3}	1.36×10^{-2}	5.17×10^{-1}	5.17×10^{-1}	5.17×10^{-3}	8.27×10^{-6}	7.03×10^{-5}	1.039
CPP-118 PEW waste accumulating in the WG/WH tank vault	PEW solutions	757 L (200 gal)	100	100	1	1.60×10^{-3}	1.36×10^{-2}	7.57×10^{-2}	7.57×10^{-2}	7.21×10^{-6}	1.21×10^{-6}	1.03×10^{-5}	0.152
CPP-119 P-cell wall drain for second-cycle P-110 condenser	Second-cycle product evaporator condensate	34,000 L (9,000 gal)	1.19×10^{-4}	8.47×10^{-5}	2.55×10^{-7}	2.73×10^{-5}	2.73×10^{-4}	4.05×10^{-6}	2.88×10^{-6}	8.66×10^{-9}	9.30×10^{-7}	9.29×10^{-6}	1.716×10^{-5}
CPP-120 CPP-601 west vent tunnel drains	Dissolver product	10 L (5.3 gal)	3.50×10^{6}	3.50×10^{6}	1.41×10^{4}	7.91×10^{-1}	4.76×10^{2}	35	35	1.41	7.91×10^{-6}	4.76×10^{-3}	70.15
	Decontamination liquid	120 L (31.6 gal)	3.50×10^{4}	3.50×10^{4}	1.41×10^{2}	7.91×10^{-3}	4.76	4.2	4.2	1.69×10^{-2}	9.49×10^{-7}	5.71×10^{-4}	8.42
CPP-121 M-cell floor liner	Third-cycle product	1 L (0.25 gal)	1.00×10^{-1}	1.00×10^{-1}	—a	—a	—a	1.00×10^{-7}	1.00×10^{-7}	—a	—a	—a	2.00×10^{-7}
CPP-122 E-cell floor liner pin holes	Dissolver product solution	11 L (3 gal)	3.50×10^{6}	3.50×10^{6}	1.41×10^{4}	7.91×10^{-1}	4.76×10^{2}	38.5	38.5	0.16	8.7×10^{-6}	5.24×10^{-3}	77.16
CPP-123 F-cell sump	E-DOG scrubber solution	285 L (75 gal)	3.50×10^{5}	3.50×10^{5}	1.41×10^{3}	7.91×10^{-2}	4.76×10	100	100	4.00×10^{-1}	1.00×10^{-4}	1.00×10^{-2}	200

a. Not a contaminant in the source term for that site.

7. REFERENCES

42 USC § 6901 et seq., 1976, "Resource Conservation and Recovery Act of 1976," *United States Code*, October 21, 1976. (The 1980 Amendment is cited as the "Solid Waste Disposal Act Amendments of 1980.")

Croff, A. G., 1980, *ORIGEN2-A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

DOE-ID, 1997, *Comprehensive RI/FS for the Idaho Chemical Processing Plant OU 3-13 at the INEEL— Part B, FS Report (Final),* DOE/ID-10572, Rev. 0, U.S. Department of Energy Idaho Operations Office, November 1997.

DOE-ID, 1999, *Final Record of Decision Idaho Nuclear Technology and Engineering Center Operable Unit 3-13*, DOE/ID-10660, Rev. 0, U.S. Department of Energy Idaho Operations Office, October 1999.

EDF-4366, 2004, "Criticality Safety and Special Nuclear Material Issues Relating to Uranium Hold-up in CPP-601," Rev. 2, Idaho Completion Project, Idaho National Engineering and Environmental Laboratory, June 2004.

EDF-5318, 2004, "Assessment of Radioactivity in INTEC Soil Contamination Site CPP-28," Rev. 0, Idaho Completion Project, Idaho National Engineering and Environmental Laboratory, November 2004.

EDF-5441, 2005, "CPP-601 Tank System Characterization," Rev. 1, Idaho Completion Project, Idaho National Laboratory, April 2005.

HWMA, 1983, "Hazardous Waste Management Act of 1983," Idaho Code § 39-4401 et seq., State of Idaho, Boise, Idaho.

INEEL, 1999, *Process Description and Operating History for the CPP-601/-640/-627 Fuel Reprocessing Complex at the Idaho National Engineering and Environmental Laboratory*, INEEL/EXT-99-00400, Idaho National Engineering and Environmental Laboratory, June 1999.

INEEL, 2002, *Voluntary Consent Order SITE-TANK-005, Combined System Identification and Characterization for the Uranium Dissolution and Extraction Process at the Idaho Nuclear Technology and Engineering Center*, INEEL/EXT-2001-00225, Rev. 2, Idaho National Engineering and Environmental Laboratory, February 2002.

Zohner, S. K., 1996, *Characterization of Nuclear Decontamination Solutions at the Idaho Chemical Processing Plant from 1982 to 1990,* INEL-96/0014, Idaho National Engineering Laboratory, March 1996.

Appendix A

Interoffice Memorandum on Radiological Source Term for the Contamination Sites CPP-80, CPP-119, and CPP-121
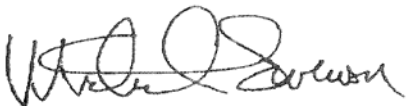
Idaho Cleanup Project

CH2M·WG IDAHO, LLC

INTEROFFICE MEMORANDUM

Date:	February 14, 2006		
To:	H. S. Forsythe	MS 3921	6-1603
From:	M. C. Swenson	MS 3950	6-3576
Subject:	RADIOLOGICAL SOURCE TERM FOR THE CONTAMINATION SITES CPP-80, CPP-119, AND CPP-121 LOCATED BENEATH BUILDING CPP-601		

References: (a) B. H. O'Brien, "Corrections to Flowsheet Calculations of Fission Product Concentrations in 2nd and 3rd Cycle Pew Streams", Internal Office Memorandum BHO-5-85, May 16, 1985

(b) M. C. Swenson, "Causes, Compositions, and Volumes of Waste Released at the INTEC Soil Contamination Site CPP-58", Internal Office Memorandum MCS-9-05, October 12, 2005

A report (ICP/EXT-05-01054) is being prepared to document the nature and extent of historical waste releases from the INTEC Fuel Reprocessing Complex (FRC), which includes buildings CPP-601, CPP-602, CPP-627, and CPP-640. This report provides radiological source term information for the three waste releases that created contamination sites CPP-119, CPP-121, and CPP-80, which are included in the FRC report. These three sites have little or no directly applicable sample data to use in estimating a source term for the waste that leaked to the soil. This report estimates the composition of the solution that leaked at those sited based upon limited sample data, fuel reprocessing flowsheet data, and comparison with similar wastes from other INTEC sources. This report provides source terms for five radionuclides of interest and nitrate for the waste that leaked. The radionuclides include Cs-137, Sr-90, H-3, I-129 and Tc-99.

Contamination Site CPP-119

Contamination site CPP-119 is located under the floor of P-cell in building CPP-601. P-cell housed the second-cycle uranium solution product evaporator (P-110). The condensate from the P-110 evaporator went to the Process Equipment Waste (PEW) collection tanks via an underground stainless steel line. In 1989 a borescope camera inspection of the condensate line was performed as part of a comprehensive evaluation of RCRA waste lines at INTEC. The inspection found a piping joint in a vertical portion of the line that had not been fully welded during the initial construction of the facility. The P-110 condensate line saw intermittent service for approximately 36 years (1953 through 1989). During that time, some waste likely leaked to the environment through the faulty weld. A draft version of the fuel reprocessing complex

H. S. Forsythe
February 14, 2006
Page 2

(FRC) report estimates 3% of the total P-110 condensate volume (about 34,000 L) may have leaked to the environment. Additional details of the leak are found in the FRC report.

The bulk of the radioactivity originally in the fuel that was reprocessed at INTEC was contained in the first-cycle raffinate. The second-cycle product had been through two cycles of uranium extraction and purification and was comparatively low in radioactivity. The activity of the condensate produced when the second-cycle product was concentrated was even lower in activity. Thus the waste that may have leaked into the soil at CPP-119 had very low activity.

The original P-110 condensate line was not equipped with any sample device, and there are no routine historical sample data of the waste. A sampler was installed in the mid 1980s and used to obtain a few (2 or 3) samples. However, the radionuclide data from those samples are very limited and their accuracy is suspect due to data inconsistencies. Therefore, this report uses historical flowsheet information (Reference 1) to estimate the radionuclide activity in the waste.

When reprocessing fuel, flowsheets (mass, energy, and material balances) were developed for each of the uranium extraction and purification cycles based upon process chemistry. The flowsheets included a diagram of the major process vessels and their inputs and outputs. The flowsheets included process operating temperatures, pressures, and compositions of the various process streams. This report uses a second/third-cycle flow sheet from 1985 as the basis for the composition of the P-110 condensate. Table 1 shows the flowsheet data used to calculate the activity of the five radionuclides of interest (and nitrate) in the release of 34,000 L of P-110 condensate at site CPP-119.

Table 1. Contaminants released to CPP-119 (34,000 L of P-110 condensate)

Nuclide or nitrate	Molecular Weight	Specific Activity (Ci/g)	Concentration in P-110 condensate (M)*	Activity in P-110 condensate (microCi/L)	Activity in 34,000 L of P-110 condensate (microCi)
Cs-137	137	86.98	1.00E-14	1.19E-04	4.05E+00
Sr-90	90	136.4	6.90E-15	8.47E-05	2.88E+00
I-129	129	1.765E-4	1.20E-09	2.73E-05	9.29E-01
H-3	3	9650	8.80E-18	2.55E-07	8.66E-03
Tc-99	99	0.01695	no estimate	2.73E-04**	9.29E+00
NO3	62	NA	2.30E-04	0.01426 g/L	0.48484 Kg

* Molar concentration of species from letter BHO-5-85 (Ref 1)
**Activity of Tc-99 assumed to be ten times that of I-129, similar to PEW Evaporator condensate per MCS-09-05

Columns 1 and 2 in Table 1 contain the data necessary to convert the molar concentration of the radionuclides (flowsheet units) into activity per unit volume. Column 3 contains the molar concentrations of the species of interest from the second-cycle flowsheet in Reference 1. Column 4 provides the activity per unit volume (product of columns 1, 2, and 3 divided by one million) of the radionuclides in column 3. Column 5 provides the estimate of the total activity released at CPP-119 in 34,000 L of waste. Reference 1 did not include an estimate of the Tc-99

A-4

H. S. Forsythe
February 14, 2006
Page 3

activity. The activity of Tc-99 was estimated as ten times that of I-129. This was based upon the similarity of the type of waste (evaporator condensate) to that of the PEW Evaporator for which a similar waste composition estimate was made (Reference 2).

Table 1 shows the activity that leaked to CPP-119 was very small, totaling only a few microCuries. The CPP-119 activity estimate includes assumptions and engineering estimates for both the amount and composition of waste. However, even if the assumptions and resulting activity estimate is in error by an order of magnitude, its impact on the total activity in contaminated INTEC soils and the resulting threat to the aquifer is negligible due to much higher activity releases elsewhere, such as site CPP-31 where tens of thousands of Curies of activity were released.

Contamination Site CPP-121

Contamination site CPP-121 is located under the floor of M-cell in building CPP-601. M-cell stored uranium product solution in tanks that were used for uranium accountability measurements. M-cell was equipped with a stainless steel liner, but a criticality safety test performed in 1983 found a small leak in the liner, the exact location of which was never identified. There were occasional leaks from pumps located within M-cell. The leaks were typically very small in volume (less than a gallon) and were quickly cleaned up. Because the cell was used for uranium accountability samples, any large leaks or losses of solution would have been noticed due to the accompanying loss of uranium. It is not known if any of the solution that leaked from the pumps into the cell subsequently leaked through cell liner. If such a leak occurred, the solution could have subsequently leaked through the joints between the concrete floor and walls of the cell and entered the surrounding soil.

This report assumes the solution that may have leaked from M-cell was third cycle product solution from the Q-cell product evaporator (Q-110). A draft version of the FRC report estimates 1 L of solution may have leaked through the liner and into the environment. Additional details of the leak are found in the FRC report.

The bulk of the radioactivity originally in the fuel reprocessed at INTEC was contained in the first-cycle raffinate. The third-cycle product had been through three cycles of uranium extraction and purification and was very low in radioactivity. Because of the low activity and small potential leakage of the M-cell solution, very little radioactivity could have leaked into the soil at CPP-121.

Fission product sample data of the M-cell solution are not readily available. Therefore, this report used historical flowsheet information (Reference 1) to estimate the radionuclide activity in the M-cell solution. Flowsheets (mass, energy, and material balances) were developed for each of the uranium extraction and purification cycles based upon process chemistry. The flowsheets included a diagram of the major process vessels and their inputs and outputs. The flowsheets included process operating temperatures, pressures, and compositions of the various process streams.

A-5

H. S. Forsythe
February 14, 2006
Page 4

Table 1 shows the flowsheet data used to calculate the activity of five radionuclides of interest (and nitrate) in the release of 1 L of M-cell solution, based upon Q-110 product. Columns 1 and 2 contain the data necessary to convert the molar concentration of the radionuclides (units used in the flowsheet) into activity per unit volume. The flowsheet concentration units (molar) of the species of interest are given in column 3. Column 4 provides the activity per unit volume (product of columns 1, 2, and 3 divided by one million) of the radionuclides in column 3. Column 5 provides the estimate of the activity released at CPP-121 in 1 L of solution. Reference 1 was the basis for the composition of all species in the Q-110 product solution except for Tc-99, for which it had no estimate. The activity of Tc-99 was estimated based upon fission yield and its ratio to Cs-137. This method has been used to estimate Tc-99 in other wastes (such as historical Tank Farm wastes) for which sample data do not exist.

Table 2. Contaminants released to CPP-121 (1 L of M-cell product solution)

Nuclide	Molecular Weight	Specific Activity (Ci/g)	Concentration in Q-110 product (M)*	Activity in Q-110 product (microCi/L)	Activity in 1 L of Q-110 product (microCi)
Cs-137	137	86.98	2.10E-10	2.50E+00	2.50E+00
Sr-90	90	136.4	4.50E-12	5.52E-02	5.52E-02
I-129	129	1.765E-4	3.10E-15	7.06E-11	7.06E-11
H-3	3	9650	6.70E-22	1.94E-11	1.94E-11
Tc-99	99	0.01695	no estimate	4.40E-04**	4.40E-04
NO3	62	NA	1.40E-01	8.68 g/L	0.00868 Kg

* Molar concentration of species from letter BHO-5-85
**Activity of Tc-99 assumed to be 1.76E-4 times that of Cs-137 based upon fission yield data.

Table 1 shows the total amount of activity that may have leaked to CPP-121 was very small, only a few microCuries. The CPP-121 activity estimate included assumptions and engineering estimates for both the amount and composition of waste released. Even if the assumptions are in error by an order of magnitude, the impact on the total activity in contaminated INTEC soils and the resulting threat to the aquifer is negligible due to much higher releases elsewhere, such as CPP-31 where tens of thousands of Curies of activity were released.

Contamination Site CPP-80

The off gas from most of the fuel reprocessing vessels (with the exception of the dissolvers) in CPP-601 ultimately joined in a common pipe that went to condenser VT-300 before exiting CPP-601. The off gas came from extraction columns, storage vessels, airlifts, air sparges, etc. The condenser removed mists and vapors before sending the off gas to the vessel off-gas treatment system in CPP-604. The condensate formed in the VT-300 condenser originally drained to the PEW collection tanks in CPP-601 via a stainless steel line. In the early 1980s the condensate drain was rerouted into an underground cast iron line. The acidic condensate corroded the cast iron drain line resulting in a release of condensate to the soil at site CPP-80. Failure of the condensate line could have occurred as early as January 1982. Leakage to the soil occurred until July 20, 1989, when the failure was discovered, a period of time not exceeding 7.6 years.

The amount of waste that leaked to the soil is unknown. This report uses the draft FRC value for the leakage of 6 L per hour for the entire 7.6 year potential leak duration (399,460 L). The draft FRC describes this as a reasonable "bounding" value for the leak. The VT-300 condensate line had no sampling capability, so there are no historical sample data of the waste. This report developed a waste source term based upon sample analyses of similar INTEC wastes (off-gas condensates) and process knowledge.

Off gas from the VT-300 condenser went to the vessel off-gas treatment system in CPP-604 and from there to the Atmospheric Protection System (APS) in CPP-649. Along the way it mixed with off-gas from the calciners, PEW Evaporator, and Tank Farm. The combined off gas went through an APS condenser (OGF-104) in CPP-649. Samples of OGF-104 condensate were taken in the early 1980s. The off-gas inputs to the APS condenser were similar and likely had similar compositions to the CPP-601 off gas and the VT-300 condenser (especially when the calciners were not operating). Therefore, the APS condensate should have been similar in composition to that of VT-300 condensate. Accordingly, the APS condensate samples were used as a basis for estimating the composition of the VT-300 condensate. The estimated compositions were compared against similar historical data (such as PEW Evaporator condensate) for reasonableness. Table 3 provides the estimates of the composition of the VT-300 condensate and the contaminants released at site CPP-80, assuming a release of 399,460 L.

Table 3. Activity released to CPP-80 (399,460 L of VT-300 condensate)

Nuclide	H-3	I-129	Sr-90	Cs-137	Tc-99	NO3
Activity in VT-300 Waste (Ci/L)	1.33E-06	1.33E-10	1.33E-07	1.33E-07	1.33E-09	0.3 M
Activity in 399,460 L of VT-300 Waste (Ci)	5.31E-01	5.31E-05	5.31E-02	5.31E-02	5.31E-04	7,400 Kg

Attachment A provides the step-by-step process by which the CPP-80 source term estimates in Table 3 were made. In general, APS condensate sample data obtained during fuel reprocessing operations were used to estimate the VT-300 condensate I-129 activity. The ratio of the I-129 activity in the condensate to that in the service waste during fuel reprocessing was then determined. The I-129 activity released to CPP-80 was estimated by applying the condensate: SW I-129 ratio to the I-129 released to SW during the entire 7.6-year leak period. The remaining radionuclides were estimated based upon their ratios to the I-129 activity, based upon the APS condensate samples. The nitrate also was based upon samples from the APS condenser.

The radionuclide estimates in Table 3 appear to be reasonable. Two of the assumptions in the CPP-80 activity release likely cause the Table 3 estimates to be conservatively high. The first is the assumption of a 6 L/hr leak for the entire duration of the potential leak period. The draft FRC describes this leak rate as a conservatively bounding assumption. The second is the estimate of the I-129 activity in the VT-300 condensate is based upon APS condensate data. The APS condensate data include times when the Calciner was operating, and when both the Calciner and fuel reprocessing were in operation. The I-129 activity in the APS condensate was lower when both fuel reprocessing and the Calciner were operating than when just the Calciner was

operating. This suggests the Calciner was the major contributor of the I-129 in the APS condensate, and the condensate from fuel reprocessing "diluted" the I-129 in the APS condensate. However, no reduction of the I-129 activity was made to account for the mixture of high-activity Calciner condensate and low-activity fuel reprocessing condensate.

Although the estimated volume of waste that could have been released at CPP-80 is large, the total radioactivity released is relatively low compared to other INTEC release sites, such as CPP-31 where tens of thousands of Curies were released. This is because the waste released at CPP-80 was off-gas condensate, which is generally low in activity for most radionuclides.

MCS:kw

Attachment

cc: D. R. Moser, MS 5111
F. S. Ward, MS 5111
J. R. Forbes, MS 3921
L. C. Tuott, MS 5310
M. W. Patterson, MS 3950
WCB DRSC, MS 3106
M. C. Swenson Letter File (MCS-01-06)

Uniform File Code: 6150
Disposition Authority: ENV1-j-1
Retention Schedule: Cutoff at the end of the program. Destroy 75 years after cutoff.

NOTE: Original disposition authority, retention schedule, and Uniform Filing Code applied by the sender may not be appropriate for all recipients. Make adjustments as needed.

Attachment A

Derivation of VT-300 condensate composition from APS condensate sample data

Table A. Summary of APS Condenser (OGF-104) condensate sample data.

Log Number	H-3 (microCi/ml)	I-129 (microCi/ml)	H-3/I-129 ratio	Sr-90 (microCi/ml)	Sr-90/I-129 ratio	Cs-137 (microCi/ml)	Cs-134 (microCi/ml)	Cs-137/134 ratio	Cs-137/I-129 ratio	NO3 (M)	Processes in Operation
810417-23	0.423	2.10E-06	2.01E+05	0.007	3.33E+04	0.52	0.01	52.0	2.48E+05	0.03	Fuel Reprocessing (Co) and WC-114
811216-15	0.15	3.70E-05	4.05E+03	1.59E-03	4.30E+01	1.02E-03	1.70E-04	6.0	2.76E+01	0.94	WC-114 Evaporator
820909-31	5.90E-05	3.70E-06	1.59E+01			5.00E-03	7.60E-05	65.8	1.35E+03	0.33	NWCF Calciner
820913-13	0.119	9.60E-06	1.24E+04	1.90E-03	5.14E+02	7.00E-03	8.10E-05	86.4	7.29E+02	0.34	NWCF Calciner
820915-30		5.90E-06		1.60E-03	1.67E+02	1.50E-03	1.80E-05	83.3	2.54E+02	0.20	NWCF Calciner
821019-10	0.252	6.30E-07	4.00E+05	4.40E-04	6.98E+02	1.35E-03	1.00E-04	13.5	2.14E+03	0.36	Fuel Reprocessing (Al) and NWCF

Shaded cells have values significantly different from other, similar samples and appear to have incorrect data.

i. Table A above shows APS condensate data from various times; some when the Calciner and fuel reprocessing were in operation and some when just the Calciner (or Calciner Evaporator) were operating. The samples taken during fuel reprocessing had lower I-129 activity than other samples. Thus the fuel reprocessing condensate is likely lower in I-129 than average APS condensate. I-129 in VT-300 condensate estimated as the average of the two OGF-104 samples taken during fuel reprocessing (first and last samples in Table A above).

$$\text{Average I-129} = (2.1E\text{-}6 + 6.3E\text{-}7)/2 \text{ microCi/ml}$$
$$= 1.37E\text{-}6 \text{ microCi/ml (or mCi/L)}$$

2. Activity of I-129 in one month's leakage of VT-300 condensate calculated assuming condensate was generated at 6 L/hr, 24 hr/day, 30 days/mo (4320 L of condensate in a month). Assumed I-129 = 1.37E-6 mCi/L (from step 1).

$$I\text{-}129 \text{ mCi/mo} = 4320 \text{ L/mo} \times 1.37E\text{-}6 \text{ mCi/L}$$
$$= 5.9 E\text{-}3 \text{ mCi/mo}$$

3. I-129 activity in VT-300 condensate will be high during fuel dissolution and first cycle extraction due to processing solution with high I-129 activity, and low during decontamination, 2/3 cycle, or process down time due to processing low activity solutions. The same is true of I-129 in the service waste (SW), it will be high during fuel dissolution and first-cycle operation. Thus there is a correlation between the I-129 in SW and the I-129 in the VT-300 condensate; both will be high during fuel dissolution and first cycle extraction, and low at other times. Calculate the VT-300:SW ratio of I-129 during fuel reprocessing. During the mid 1980s, the I-129 in SW averaged about 5 mCi/month during fuel reprocessing. VT-300 I-129 was about 0.0059 mCi/month (from step 2).

$$VT\text{-}300:SW \text{ I-129 ratio} = 0.0059/5$$
$$= 0.00118$$

4. The amount of I-129 released to SW varied significantly during the 6.7 years of potential VT-300 leakage. The monthly releases tracked closely the operation of the Calciner, WC-114 evaporator, and fuel reprocessing. The I-129 releases were relatively high during those operations and negligible at other times. Assume the I-129 from fuel reprocessing was 1/3 of the total I-129 in the service waste (the remaining 2/3 coming from the Calciner and WC-114 evaporator). The total I-129 released to SW during the 6.7 year VT-300 potential condensate leak period was about 136 mCi. Therefore, about 45 mCi are assumed to have come from fuel reprocessing. Assume the VT-300:SW ratio of I-129 calculated in step 3 (0.00118) applied to the I-129 released from fuel reprocessing to SW is applicable to the 45 mCi I I-129 released during the VT-300 leak period. Then the I-129 in the VT-300 condensate that leaked is:

$$I\text{-}129 \text{ in VT-300} = (45 \text{ mCi}) \times 0.00118$$
$$= 0.0531 \text{ mCi}$$

5. Calculate average I-129 activity in 399,460 L of VT-300 condensate (volume from draft FRC). Total I-129 activity = 0.0531 mCi from step 4.

$$Average \text{ I-129} = 0.0531 \text{ mCi} / 399,460 \text{ L}$$
$$= 1.33E\text{-}7 \text{ mCi/L}$$
$$= 1.33 E\text{-}10 \text{ Ci/L}$$

6. Determine activity of other radionuclides base upon evaluation of activity in Table A (ratio to activity of I-129), engineering judgment, and comparison to other similar wastes (evaporator condensate).

   A. Cs-137:I-129 ratio is about 1,000:1   Sr-90:I-129 ratio is about 1,000:1   H-3:I-129 ration is about 10,000:1
      Note the ratios come from Table A. Ratios vary, but selected values are within the data range and comparable with PEW Evaporator condensate data.

   B. Tc-99 There are no sample data available. Assume Tc-99 is 10 times the I-129 activity based upon other similar (PEW Evaporator condensate) source term evaluations (MCS-9-05).

   C. NO_3 concentration is about 0.3 molar.

7. Calculate activity/concentration of species in VT-300 condensate using I-129 = 1.33E-10 Ci/L (from step 5) and ratios from step 6. Then multiply each value by 399,460 L to calculate the amount released by VT-300 into CPP-80. Include this data in Table 3 of report text.

   A. Cs-137 = 1.33E-10 x 1,000 = 1.33E-7 Ci/L   Cs-137 in 399,460 L of waste Cs-137 = 0.0531 Ci
   B. Sr-90 = 1.33E-10 x 1,000 = 1.33E-7 Ci/L   Sr-90 in 399,460 L of waste Sr-90 = 0.0531 Ci
   C. H-3   = 1.33E-10 x 10,000 = 1.33E-6 Ci/L   H-3 in 399,460 L of waste H-3 = 0.531 Ci
   D. Tc-99 = 1.33E-10 x 10 = 1.33E-9 Ci/L   Tc-99 in 399,460 L of waste Cs-137 = 5.31 E-4 Ci

A-11

1. DOCUMENT/ RECORD INFORMATION

Document ID: MCS-01-06   Revision ID: 0   Project Number: N/A

Title/Description: CPP-601 Radiological source Term for the Contamination Sites CPP-80, CPP-119, and CPP-121 Located Beneath Building

Alternate ID: MCS-01-06   Date of Record: OR Effective Date: 2-4-2006

Document Author(s)/Creator(s): Michael C. Swenson   Date Range From: _____ (if applicable):

Document Owner: Michael C. Swenson   To: _____

Originating Function: Calcine Disposition   Charge No.: P10666114

Controlled Documents Only:
Periodic Review Frequency: ☒ N/A   ☐ 5 years   ☐ Annually   ☐ Other: (Explain) _____

2. RECORDS MANAGEMENT REQUIREMENTS

Uniform File Code(s): 6150   Disposition Authority: ENV1-j-1

Keywords: Release, Leak, Spill, Release, Fuel Reprocessing

Do record(s) pertain to: ☐ DOE/RW-0333P   ☐ WIPP/TRU Waste   ☐ NRC Funded programs
If pertinent, provide Total Pages: _____   OR ☐ N/A if bound (see instructions)

Choose applicable **FACILITY** that record(s) pertains to:
☐ ARA   ☐ CFA   ☐ EBR-1   ☐ IF FACILITIES   ☒ INTEC   ☐ MFC (formerly ANL-W)
☐ NRF   ☐ PBF/WERF/WROC ☐ RTC (formerly TRA) ☐ RWMC   ☐ SMC   ☐ TAN
☐ Site-Wide ☐ OTHER   (If Site-Wide or OTHER, add facility names in keywords.)

Category: ☐ Quality Assurance   ☐ Sensitive Unclassified   ☐ Emergency Operating   ☐ Legal & Financial Rights
☒ Environmental **(complete shaded area below)**

ENV. RECORD INFORMATION	2a. CERCLA	☒ Yes ☐ No	If YES, provide:	WAG **3**	OU **13**	SITE CODE **80, 119, 121**		
	2b. Env. Applicability: (Check one)	AIR:	☐ Site Wide General		☐ Site Wide NOx	☐ SW Title V		
		CULTURAL RESOURCES:	☐ Archaeological	☐ Architectural	☐ Archival			
		HAZ/RCRA:	☐ Closure	☐ TSDF	☒ Waste Analysis/Characterization		☐ Other	
		WATER:	☐ Drinking	☐ Storm	☐ WLAP	☐ Wells	☐ Other	
		☐ EPCRA	☐ FIFRA	☐ NEPA	☐ TSCA	☐ MISCELLANEOUS		

3. QA RECORD VALIDATION (QUAL: QLRMVVSW)
Only required for QA records where no other method of validation (signature, initials, etc.) is available on the record itself

Validator Print/Type Name	Validator Signature	Validator S Number	Date

4. SIGNATURES

Sign/Date as Sender: _____ Michael C. Swenson _____   *[signature]*   3-22-200
Sender Print/Type Name   Sender Signature   Date

Acceptance/ Receipt: _____ TSB _____   *[signature] Carly Anderson*   3/20/06
Location of Receiving DRSC or SFL   Receiver Signature   Date

5. RECORDS PROCESSING INFORMATION
For Document Control and Records Management Use Only

Image ☑   Vault Page ☐   *CA*   Import *CA*   Index *CA*   QC _____

Appendix B

Interoffice Memoranda on Chemical Composition of PEW Evaporator Feed Solutions and Evaporator Bottoms

EXXON NUCLEAR IDAHO COMPANY, Inc.

Internal Correspondence

Date: February 17, 1981

To: W. B. Palmer W6P

From: D. W. Rhodes

Subject: Composition of PEW Evaporator Feed, Condensate and Bottoms/Rhod-1-81

Distribution:
R. D. Bradley
D. B. Chamberlain
B. R. Dickey
S. J. Fernandez
G. E. Lohse
G. J. McManus
L. W. McClure
W. A. Mickelson
E. P. Mondok
L. F. Morrow
B. E. Paige
G. L. Ritter

R. E. Schindler
D. W. Siddoway
S. G. Smolen
M. D. Staiger
F. S. Ward
D. K. Wenzel

During the period January 17 through January 21, 1981, an extensive sampling and analysis program was undertaken in the PEW evaporator system, at your request. The purpose of the program was to determine the composition of the feed solution and the condensate and bottoms resulting from boil down of this feed solution. During this sampling period, the WCF was operating on waste solution from tanks WM-185 and WM-189, and some decontamination was underway. The fuel dissolution and extraction processes were not operating.

The feed solution to the evaporator consisted of waste collected in, WG-100, WH-100, WG-101 and WH-101 (deep tanks); OGF-104 (APS condenser); and SFE-106 (fuel storage basin sludge collection tank). Each vessel was sampled. The evaporator bottoms sample was taken from WL-613, and the condensate samples were taken from WL-106 and WL-107 evaporator condensate collection tanks.

The results of the chemical and radiochemical analyses as well as the date sampled and the volume of the waste in each vessel are given in tables 1 through 4.

A similar sampling and analysis program is planned for the future when the fuel processing equipment is operating and the WCF is shut down.

If you have any questions, please contact me at 6-3080.

Attachments

dg

TABLE 1

WCF ↑
Fuel ↓

CHEMICAL COMPOSITION OF PEW EVAPORATOR FEED
SOLUTIONS AND EVAPORATOR BOTTOMS

are (PEW/CFA TANKS)

bottoms

COMPONENT	UNITS	WG-100	WH-100	WG-101	WH-101	OGF-104	SEF-106	WL-613
Cl	mg/L	67.7	65.3	129.2	37.0	6.9	187.5	707.3
F	mg/L	0.74	15.1	4.5	87.3	12	<8.4	18.2
SO$_4$	mg/L	28	36	38	65	<5	---	4501
Hg	mg/L	0.02	0.013	0.02	0.02	36		339
Al	mg/L	3	21	16	50	0.9	20	3500
B	mg/L	---	1	< 0.5	---	8	---	144
Ca	mg/L	3	8	4	45	7	44	2500
Cr	mg/L	2	1	6	---	0.3	---	410
Cu	mg/L	0.1	2	8	7	---	0.2	100
Fe	mg/L	7	11	9	---	1	5	930
Mg	mg/L	3	3	3	17	1	10	380
Mn	mg/L	3	1	25	---	---	0.4	15
Na	mg/L	468	557	2086	337	---	699	34900
Ni	mg/L	0.7	0.6	0.4	---	0.3	0.5	150
Pb	mg/L	6	53	8	---	---	---	1004
Si	mg/L	8	10	12	10	11	20	10
Sn	mg/L	---	---	7	---	---	---	550
Zr	mg/L	2	4	1	---	---	---	1453
K	mg/L	---	---	---	---	---		16500
Mo	mg/L	---	---	2	---	---		---
PO$_4$	mg/L	---	---	---	---	---		960
NO$_3$	mg/L	2.9×10^4	9.1×10^3	1.6×10^4	1.7×10^4	2.8×10^4	3.2×10^3	2.7×10^5
pH		---	---	0.88	0.73	---		---
Acid	N	0.230	0.147			0.431	---	0.617
UDS	mg/L	---	---	---	---	---	---	6690
Sp.Gr.		---	---	---	---	---	---	1.23
Date		1/17/81	1/18/81	1/20/81	1/21/81	1/19/81	1/20/81	1/21/81
Volume Gal.		2500	2750	3000	3000	5750	3200	250

TABLE 2

RADIOCHEMICAL COMPOSITION OF PEW EVAPORATOR

FEED SOLUTIONS AND EVAPORATOR BOTTOMS

COMPONENT	UNITS	VESSEL NUMBER						
		WG-100	WH-100	WG-101	WH-101	OGF-104	SFE-106	WL-613
U	mg/L	0.901	4.91	—	0.144	—	—	72.3
Pu	μCi/mL	8.2×10^{-4}	1.3×10^{-3}	4.8×10^{-4}	6.4×10^{-4}	2.1×10^{-6}	1.4×10^{-5}	—
ross B[1]	μCi/mL	1.1×10^{-2}	1.20	3.9×10^{-2}	0.98	0.65	—	80.1
e-144	μCi/mL	1.1×10^{-2}	4.1×10^{-2}	2.4×10^{-3}	1.3×10^{-2}	2.4×10^{-3}	—	3.55
o-60	μCi/mL	1.7×10^{-4}	0.11	3.0×10^{-4}	2.3×10^{-3}	—	1.7×10^{-4}	0.19
s-134	μCi/mL	6.0×10^{-3}	3.4×10^{-2}	1.8×10^{-3}	8.4×10^{-3}	—	3.3×10^{-5}	2.63
s-137	μCi/mL	2.4×10^{-2}	0.21	9.6×10^{-3}	3.9×10^{-2}	2.3×10^{-2}	9.4×10^{-4}	12.2
u-152	μCi/mL	—	—	—	—	—	8.78×10^{-3}	0.21
u-154	μCi/mL	4.5×10^{-4}	3.8×10^{-3}	1.3×10^{-4}	5.0×10^{-4}	—	6.8×10^{-3}	0.38
u-155	μCi/mL	2.4×10^{-4}	8.0×10^{-3}	6.8×10^{-5}	—	—	1.3×10^{-3}	0.18
u-106	μCi/mL	7.8×10^{-3}	0.91	3.8×10^{-3}	0.13	0.50	—	20.3
r-90	μCi/mL	2.2×10^{-2}	0.24	8.2×10^{-3}	3.1×10^{-2}	1.1×10^{-4}	4.6×10^{-3}	0.11
b-125	μCi/mL	3.2×10^{-3}	0.17	6.2×10^{-3}	1.4×10^{-2}	—	—	0.45
r-144	μCi/mL	1.1×10^{-2}	4.1×10^{-2}	2.4×10^{-3}	1.3×10^{-2}	—	2.4×10^{-3}	3.55
-90	μCi/mL	2.2×10^{-2}	0.24	8.2×10^{-3}	3.1×10^{-2}	1.1×10^{-4}	4.6×10^{-3}	0.11
n-54	μCi/mL	—	9.4×10^{-4}	—	—	—	—	7.4×10^{-2}
-3	μCi/mL	—	—	—	—	0.12	—	—
-129	μCi/mL	$<6.0 \times 10^{-5}$	$<2.9 \times 10^{-4}$	$<3.0 \times 10^{-6}$	$<4.2 \times 10^{-6}$	1.7×10^{-5}	—	$<2.4 \times 10^{-}$
DATE		1/17/81	1/18/81	1/20/81	1/21/81	1/19/81	1/20/81	1/21/81
VOLUME	GAL	2600	2750	3000	3000	5760	3200	250

1) DOES NOT INCLUDE TRITIUM

TABLE 3

CHEMICAL COMPOSITION OF EVAPORATOR

CONDENSATE

VESSEL NUMBER

COMPONENT	UNITS	WL-106	WL-107	WL-106	WL-107	WL-106		Average
Cl	mg/L	34.6	36.4	59.1	57.8	57.3		49.0
F	mg/L	16.3	16.8	16.4	16.4	14.5		16.1
SO_4	mg/L	<5	<5	<5	20	<50		<17
NO_2	mg/L	423	445	75	182	512		434
NO_3	mg/L	1.7×10^4	1.8×10^4	1.6×10^4	1.5×10^4	1.4×10^4		1.6×10^4
Na	mg/L	0.1	<0.1	0.9	0.7	<0.1		<0.4
Hg	mg/L	4.0	5.0	5.1	5.8	5.3		5.0
pH		0.63	0.57	0.66	0.63	0.87		0.67
Acid	N	0.313	0.375	0.270	0.270	0.270		0.30
VOLUME	GAL	4000	4000	4000	4000	4000		
DATE		1/18/81	1/19/81	1/20/81	1/21/81	1/21/81		
TIME		09:56:57	02:47:29	18:03:10	05:43:56	17:43:58		

B-6

TABLE 4

RADIOCHEMICAL COMPOSITION OF EVAPORATOR

CONDENSATE

MPONENT	UNITS	WL-106	WL-107	WL-106	WL-107	WL-106		Average
CSS B[1]	µCi/mL	5.5×10^{-5}	6.3×10^{-5}	$<1.1 \times 10^{-4}$	8.4×10^{-4}	6.8×10^{-5}		4.9×10^{-5}
3	µCi/mL	4.3×10^{-2}	6.2×10^{-2}	5.2×10^{-2}	4.1×10^{-2}	4.7×10^{-2}		4.9×10^{-2}
-144	µCi/mL	3.4×10^{-6}	4.1×10^{-6}	3.7×10^{-6}	4.0×10^{-6}	3.2×10^{-6}	0.999	3.7×10^{-6}
-60	µCi/mL	1.7×10^{-7}	——	1.5×10^{-7}	1.8×10^{-7}	1.6×10^{-7}	1.11	1.7×10^{-7}
-134	µCi/mL	2.1×10^{-6}	2.3×10^{-6}	2.2×10^{-6}	2.3×10^{-6}	2.1×10^{-6}	1.145	2.2×10^{-6}
-137	µCi/mL	1.1×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.3×10^{-5}	1.1×10^{-5}	1.02	1.2×10^{-5}
-154	µCi/mL	1.9×10^{-7}	——	2.3×10^{-7}	3.1×10^{-7}	2.6×10^{-7}		2.5×10^{-7}
-155	µCi/mL	——	2.2×10^{-7}	——	2.5×10^{-7}	——	0.533	2.3×10^{-7}
-106	µCi/mL	1.3×10^{-5}	1.9×10^{-5}	2.2×10^{-5}	2.5×10^{-5}	2.4×10^{-5}		2.1×10^{-5}
-252	µCi/mL	——	——	4.2×10^{-7}	9.3×10^{-7}	3.1×10^{-7}		5.5×10^{-7}
r-90	µCi/mL	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.4×10^{-5}	0.240	1.2×10^{-5}
b-125	µCi/mL	4.7×10^{-7}	——	——	3.6×10^{-7}	——	1.07	4.2×10^{-7}
r-144	µCi/mL	3.4×10^{-6}	4.1×10^{-6}	3.7×10^{-6}	4.0×10^{-6}	3.2×10^{-6}	——	3.7×10^{-6}
-90	µCi/mL	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.4×10^{-5}	——	1.2×10^{-5}
-131	µCi/mL	1.7×10^{-7}	2.6×10^{-7}	1.7×10^{-7}	1.2×10^{-7}	——	——	1.8×10^{-7}
-129	µCi/mL	9.3×10^{-6}	1.3×10^{-5}	7.2×10^{-6}	5.1×10^{-6}	6.3×10^{-6}		5.6×10^{-6}
OLUME	GAL	4000	4000	4000	4000	4000		4000
ATE		1/19/81	1/19/81	1/20/81	1/21/81	1/21/81		
IME		09:56:57	02:47:29	18:03:10	05:43:56	17:43:58		

1) DOES NOT INCLUDE TRITIUM

B-7

435.55
01/16/2003
Rev. 01

HLW PROGRAM
RECORDS INPUT FORM

HLW-100 _1948_ TS

	Instructions: Fill in bolded fields or use N/A when applicable. See page two for instructions.	
1.	**Document ID Number and Rev:**	HLW-100-1948 / MCS-1-84
2.	**Document Date:**	02/01/1984
3.	**Document Title/Description or Subject:**	PROCESS EQUIPMENT WASTE (PEW) EVAPORATOR FEED STREAM COMPOSITION
4.	**Addressee Name and Organization:**	W.B. PALMER
5.	**Author Name and Organization:**	M.C. SWENSON
6.	**Key Words:**	PROJECT NUMBER 23582
7.	**Uniform Filing Code:** 6150	**Disposition Authority:** ☒ ENV1-j-1
		☐ Other:
	Retention Schedule: ☐ EPI Cutoff at end of the program. Destroy 75 years after cutoff. **LONG TERM STEWARDSHIP** ☒ Other: Cutoff at end of the program. Destroy 75 years after cutoff.	
8.	**Document Type:** (See instruction page.)	CORRESPONDENCE
9.	**File Location:** (indicate only one.)	☒ WCB ☐ EROB ☐ CPP-637
	Total Pages of Document: 21	☐ Satellite: ☒ Other: **SUBPROJECT 3 GENERAL**
10.	**If QA Record, Validator Sign and Date:**	Sign: _____ Date: _____
		(Mark box only if applicable) ☐ QA Document Pages _____
	QA Record Classification:	☐ Non Permanent ☐ Lifetime
11.	**Scan Document:**	☒ Scan into EDMS ☐ No Scan
		☐ Unclassified/Business Sensitive
	Medium:	☒ Hard Copy ☐ Disk ☐ Other (Specify)
12.	**Receiver/Verifier**	Sign: CATHY GONZALES Date: 02/24/04
13.	**Individual Submitting Record** (Please Print)	Name: M.D. STAIGER Date: 02/24/04

Incomplete forms will be returned to Individual Submitting Record.

B-8

435.55
01/16/2003
Rev. 01

HLW PROGRAM
RECORDS INPUT FORM

1. If the document to be submitted bears a number (report number, letter number, etc.), enter it, along with the revision status.

2. Enter document date.

3. If the document to be submitted has no title, enter a brief subject description.

4. If the document to be submitted is correspondence, enter the name of the person to whom the correspondence is addressed, and that person's organization. If the correspondence has multiple addressees, enter "Distribution."

5. Enter the name of the person who signed the letter or authored the report; if no approval signature appears on the document, enter N/A.

6. Enter key subject terms that would be useful in retrieving the document. Avoid duplicating information provided in the document title/description or subject field.

7. Identify Uniform File Code (UFC) from HLW Program Records Type List. HLW Program case file records are filed under UFC-6150.

    Mark the appropriate boxes for Disposition Authority and Retention Schedule for the document.

    Disposition Authority:

    ENV1-j-1   (HLW Program Case File) Cutoff at end of program. Destroy 75 years after cutoff.

    Other   Identify Disposition Authority and retention period as specified on the HLW Program Record Type List.

8. Document Types: (more than one may be applicable)

As-Built Drawings	Funding/Budget Documentation	Procedures
Assessments, Audits, Appraisals	Independent Hazard Review	Reference Material
Construction Documents	Justifications	Reports
Contract or Work Agreements	Logbooks	Reviews
Correspondence	Minutes/Agendas	Sampling Analysis
Data Sheets	Miscellaneous Documents	Statements of Work
Electronic Mail	Organization Structure	Training
Engineering Design Files	Permits	Waste Characterization Data
Environmental Monitoring	Plans	
Facsimile	Presentations	

9. Specify the Central File Location (CFL) or the Satellite CFL where the document will reside. Enter the total page count including two pages for Form 435.55.

10. Quality Assurance Validator signs and dates in space provided. If the document is a QA record, mark the box as indicated and enter total page count including two pages for Form 435.55. Determined by the Quality Assurance Validator.

11. Mark if the document is to be scanned into EDMS HLW Cabinet or if No Scan is required. If document should be placed in a restricted folder, mark Unclassified/Business Sensitive to limit viewing access.

12. CFL personnel will receive and verify document.

13. Individual submitting record.

EXXON NUCLEAR IDAHO COMPANY, Inc.

internal Correspondence

Distribution:
R. D. Bradley
D. B. Chamberlain
B. R. Dickey
S. J. Fernandez
G. W. Hogg
G. E. Lohse
G. J. McManus
L. W. McClure
W. A. Mickelson
E. P. Mondok
L. F. Morrow
B. E. Paige
G. L. Ritter

Date: June 15, 1981

To: W. B. Palmer/WBP

From: D. W. Rhodes/Rhod-6-81

Subject: Composition of PEW Evaporator Feed,
Condensate and Bottoms Solutions

During the period April 3 through April 8, 1981, a sampling program was completed to help define the concentrations of chemicals and radionuclides in PEW feed, condensate, and evaporator bottoms solutions while the uranium recovery process was operating, but the WCF was shut down. The results of a similar sampling program with the WCF operating and the uranium recovery process shut down were reported earlier (Rhod-1-81).

The feed solution to the evaporator during the test consisted of waste from WG-100, WH-100, WG-101 and WH-101 deep tanks; OGF-104, the APS condenser; and WC-119, the WCF sump tank. The condensate was collected in WL-106 and WL-107 tanks; the evaporator bottoms sample was taken from WL-613. The samples from each source were made into a composite, which was analyzed. Thus, the analytical results actually represent average values. The results are given in Tables 1 and 2. If you have any questions, please contact me on 6-3080.

CW

Attachment

Cont. cc: R. E. Schindler
            D. W. Siddoway
            S. G. Smolen
            M. D. Staiger
            F. S. Ward
            D. R. Wenzel

Rhod-6-81
Page 1

TABLE 1

WCF Shut Down   *reprocessing Fuel ↑ (75)*

CHEMICAL COMPOSITION OF PEW EVAPORATOR FEED SOLUTIONS,
CONDENSATE AND BOTTOMS

COMPONENT	UNITS	Comp. of Soln in PEW Evap.	WG-100-101 WH-100-101	OGF-104	WC-119	WL-106-107	WL-613
Cl	mg/L	53...	42.0	26.6	577	34.9	1001
F	mg/L	48.4 mg/L	14.0	17.2	1046	21.2	2101
SO4	mg/L		<25	28	ND	<1	ND
Hg	mg/L		0.9	3	370	ND	6840
Al	mg/L		10	12	800	4	2400
B	mg/L		ND	ND	80	ND	80
Ca	mg/L		0.8	10	400	0.4	800
Cr	mg/L		4	ND	40	ND	600
Cu	mg/L		1	ND	40	ND	32
Fe	mg/L		4	ND	40	ND	800
Mg	mg/L		4	3	80	0.05	200
Mn	mg/L		16	ND	30	ND	1000
Na	mg/L		1130	24	370	0.2	70,000
Ni	mg/L		0.6	ND	100	ND	140
Pb	mg/L		15	ND	ND	ND	700
Si	mg/L		7	6	40	ND	10
Zr	mg/L		2	ND	240	ND	12
K	mg/L		334	3	278	<0.05	16,000
NO3	M		0.24	0.03	0.58	0.29	6.63
pH			0.86	1.5	-	0.65	-
ACID	N		.1	.63M	0.42		2.1
UDS	mg/L						5.75
Sp. Gr.							1.33
Volume Gal (Liters)			27,320 (103,446)	1400 (5,???)	1000	27,800	350
Log No.			041722	041723	041724	141721	040818

ND = Not Detected

Attachment
Rhod-6-81
Page 2

040613

TABLE 2

RADIOCHEMICAL COMPOSITION OF PEW EVAPORATOR FEED, SOLUTIONS, CONDENSATE AND BOTTOMS

COMPONENT	UNITS	WG-100-101 WH-100-101	OGF-104	WC-119	WL-106-107	WL-613
				VESSEL NUMBERS		
U	mg/L	0.63	0.14	1.79	<5 E-2	70
Pu	µCi/mL	1.1 E-3	9.2 E-5	IS	2.6 E-5	0.14
I-129	µCi/mL	<1.3 E-6	2.1 E-6	IS	5.7 E-7	IS
Gross β	µCi/mL	NR	NR	NR	NR	1.2 E+4
Ce-144	µCi/mL	0.05	1.8 E-3	1.1	7.8 E-6	6.7
Co-60	µCi/mL	3.9 E-4	ND	ND	ND	0.1
Cs-134	µCi/mL	0.01	0.01	4.3	7.4 E-6	8.3
Cs-137	µCi/mL	0.11	0.52	63.3	9.1 E-5	106.3
Eu-152	µCi/mL	ND	ND	ND	ND	0.3
Eu-154	µCi/mL	1.5 E-3	2.8 E-4	0.2	ND	0.6
Eu-155	µCi/mL	6.4 E-4	ND	ND	ND	0.3
Ru-106	µCi/mL	0.04	0.01	2.5	1.0 E-5	7.4
Sr-90	µCi/mL	0.11	0.07	13.5	3.2 E-5	36.2
Sb-125	µCi/mL	0.01	ND	0.2	ND	0.4
Pr-144	µCi/mL	0.05	1.8 E-3	1.1	7.8 E-6	6.7
Y-90	µCi/mL	0.11	0.07	13.5	3.2 E-5	36.2
H-3	µCi/mL	4.4 E-4	0.42	0.05	0.14	NR
C-14	µCi/mL	7.1 E-7	1.5 E-8	IS	6.8 E-7	NR

NR = Not Requested

ND = Not Detected

IS = Insufficient Sample

EXXON NUCLEAR IDAHO COMPANY, Inc.

Internal Correspondence

Date: October 5, 1983

To: W. E. Palmer WEP

From: D. W. Rhodes/Rhod-17-83

Subject: Composition of PEW Evaporator Feed, Condensate and Bottoms

Distribution:

D. R. Alexander
B. R. Dickey
G. W. Hogg
J. E. Johnson
G. E. Lohse
L. W. McClure
W. A. Mickelson
R. E. Mizia
E. P. Mondok
P. I. Nelson
A. P. Roeh
R. E. Schindler
M. D. Staiger
M. C. Swenson
F. S. Ward
D. W. Rhodes-2

During the period August 5 through August 11, 1983, a sampling program was completed to help define the concentrations of chemicals and radionuclides in PEW feed, condensate and bottoms while the NWCF was operating.* The results of similar sampling programs were reported earlier in (1) Rhod-6-81 (uranium recovery process operating) and (2) Rhod-1-81 (WCF operating).

The feed solution to the evaporator during the test consisted of waste from tanks WG-100(2), WH-101, WC-119, NCD-123, WL-104, and SPE-105. The waste solutions from the PEW-CFD tanks, WG-100 and WH-101, were analyzed as a composite. The condensate samples from WL-107(2), WL-163, and WL-106 were also analyzed as a composite. The results are given in Tables I, II, and III. No interpretation of the results was attempted, since each user has a different need. If you have any questions, please contact me at 6-3080.

---

*The uranium extraction systems were not operating and the Rover burner operated only one day during the sampling period.

rls

Table I

Chemical Composition of PEW Evaporator
Feed Solutions, Condensate and Bottoms

Component	Unit	WG-100 WH-101	WC-119	NCD 123	WL-104	SFE-106	WL-613	WL-106 WL-107 WL-163
Cl	mg/L	67	135	97	83	195	2435	43
F		16	29	296	1424	0.9	1777	47
SO$_4$		83	12	42	104	24	6336	< 1
Hg		0.2	6.8	4E-3	0.7	2E-3	1201	7.7
Al		240	9.6	14	460	100	2700	ND
B		ND	3.9	ND	20	ND	1800	< 2
Ca		11	37	230	160	250	18000	1
Cr		ND	0.6	52	2.6	ND	580	< 0.4
Fe		ND	3.4	39	13	16	1100	0.7
Mg		19	8.4	160	55	27	1800	0.3
Mn		ND	0.3	1500	0.6	2.4	7000	ND
Mo		ND	ND	ND	6.6	ND	ND	ND
Na		620	10	6110	185	488	110000	0.2
Ni		ND	< 0.1	ND	ND	1.4	48	ND
Pb		ND	ND	21	35	ND	470	ND
Si		< 20	3.8	71	44	29	430	< 0.2
K		23	3	1927	79	8	26000	9
Ti		ND	ND	12	3.3	ND	ND	ND
Zn		ND	1	ND	ND	ND	39	ND
NO$_3$		13640	20646	5580	4588	2728	556000	5270
pH		1.38	0.77	12.5	1.48	2.34	0.55	1.23

ND = Not Detected

Table II

Sources of Waste Solutions During Boildown Test*

Date	Feed Source	Gallons	Tank	Condensate Batch No.	Gallons
7-5	WG-100	2650	WL-107	227	4000
7-6	SFE-106	1250	WL-163	228	4000
7-7	WC-119	2100	WL-106	229	4000
7-7	WH-101	2900	WL-107	230	4000
7-10	NCD-123	2700			16000
7-11	WG-100	2550			
7-11	WL-104	700	WL-613	Evaporator Bottoms	200
	Total	14850			16200

*The difference between the feed volume and the volume of condensate
plus evaporator bottoms is 16200-14850=1350 gal. This difference is
due to the input of small volumes of water from a variety of sources
(VOG loop seals, off-gas drains, pump priming, etc.), that are not
measured as well as instrument and/or reading errors.

Table III

Radiochemical Composition of PEW Evaporator
Feed Solutions, Condensate and Bottoms

Component	Unit	WG-100 WH-101	WC-119	NCD 123	WL-104	SFE-106	WL-613	WL-106 WL-107 WL-163
Total U	mg/L	2.3	<0.055	<0.055	3.9	0.22	217	<0.055
I-129	µCi/mL	6.3E-5	1.7E-4	3.1E-6	5.5E-6	2.9E-5	2.1E-4	8.2E-6
Ce-144		6.1E-3	9.8E-2	6.5E-5	ND	5.8E-4	1.2	5.4E-7
Co-60		3.2E-4	5.5E-5	6.6E-4	1.5E-4	6.3E-4	2.7E-2	ND
Cs-134		6.3E-4	1.5E-2	ND	7.1E-4	ND	0.17	2.8E-7
Cs-137		2.6E-2	1.6E-1	5.7E-4	1.9E-2	3.7E-4	2.73	5.2E-6
Eu-152		ND	ND	ND	ND	1.8E-2	0.16	ND
Eu-154		ND	1.1E-3	ND	ND	1.4E-2	0.13	ND
Eu-155		ND	6.0E-4	ND	5.9E-4	3.1E-3	4.7E-2	ND
Ru-106		0.29	1.0E-2	2.3E-3	1.9E-2	ND	3.9	4.3E-5
Sr-90		3.3E-2	0.10	7.7E-6	3.3E-3	1.2E-2	3.5	1.9E-6
Sb-125		6.1E-3	1.8E-3	5.7E-5	3.9E-2	ND	0.45	4.1E-6
Mn-54		ND	ND	ND	4.9E-5	ND	ND	ND
Pr-144		6.1E-3	9.8E-2	6.5E-5	ND	5.8E-4	1.2	5.4E-7
Y-90		3.3E-2	0.10	7.7E-6	3.3E-3	1.2E-2	3.5	1.9E-6
H-3		1.7E-3	0.26	1.5E-4	3.0E-3	3.8E-4	2.1E-2	4.0E-2
Nb-95		ND	5.6E-4	2.1E-5	ND	ND	9.7E-2	8.2E-7
K-40		ND	ND	ND	ND	1.3E-3	ND	ND
Zr-95		ND	1.0E-3	ND	ND	ND	8.6E-2	ND

ND = Not Detected

B-15

EXXON NUCLEAR IDAHO COMPANY, Inc.

Internal Correspondence

Distribution:

D. R. Alexander
B. R. Dickey
G. W. Hogg
J. E. Johnson
G. E. Lohse
L. W. McClure
W. A. Mickelson
R. E. Mizia
E. P. Mondok
P. I. Nelson
A. P. Roeh
R. E. Schindler
M. O. Staiger
F. S. Ward
D. W. Rhodes-2

Date: October 24, 1983

To: W. B. Palmer

From: D. W. Rhodes/Rhod-19-83 *DW Rhodes*

Subject: Composition of PEW Evaporator Feed, Condensate and Bottoms

Ref: Letter, D. W. Rhodes to W. B. Palmer, "Composition of PEW Evaporator Feed, Condensate and Bottoms," Rhod-17-83, October 5, 1983.

At the time the referenced letter was published, the values for plutonium were not available. These values are now available and are given below as duplicate analyses of the same sample. The duplicate values are reported to show the uncertainties involved in measuring the very low concentrations of the isotopes of plutonium. If you have any questions, call me at 6-3080.

Plutonium Isotope	Unit	WC-119	NCD-123	WL-104	SFE-106	WL-613	WL-106 WL-107 WL-163
Total	µg/g	0.026	0.016	0.012	0.001	0.571	0.0002
		0.022	0.015	0.013	0.001	0.395	0.0003
	µg/mL	0.026	0.016	0.012	0.001	0.569	0.0002
		0.022	0.015	0.013	0.001	0.394	0.0003
238	wt%	<25.0	11.5	<17.8	4.2	3.0	<50.6
		<25.2	10.5	<18.4	3.0	<25.2	<21.0
239	wt%	28.0	29.9	50.1	30.2	51.2	23.1
		28.6	28.7	48.5	31.2	57.7	31.6
240	wt%	18.5	23.4	11.7	23.2	20.3	18.9
		19.4	24.2	15.3	24.0	9.2	3.4
241	wt%	16.2	18.2	8.1	22.1	13.0	6.2
		15.9	19.8	9.4	21.9	6.2	38.1
242	wt%	12.3	17.0	12.3	20.3	12.5	1.2
		10.9	16.8	7.9	19.9	1.7	5.9

HLW-100 *1948*   **LTS**

EXXON NUCLEAR IDAHO COMPANY, Inc.

Internal Correspondence

Distribution:
R. I. Donovan
H. B. Eldredge
G. W. Hogg
R. A. Kirkbride
D. W. Rhodes
A. P. Roeh
M. D. Staiger
J. H. Valentine
F. S. Ward
M. C. Swenson-2

Date: February 1, 1984

To: W. B. Palmer/G. G. Simpson

From: M. C. Swenson/MCS-1-84

Subject: PEW Evaporator Feed Stream Composition

Ref: 1) Letter, D. W. Rhodes to W. B. Palmer, "Composition of
PEW Evaporator Feed, Condensate, and Bottoms," Rhod-1-81,
dated February 17, 1981.

2) Letter, D. W. Rhodes to W. B. Palmer, "Composition of
PEW Evaporator Feed, Condensate, and Bottoms Solutions,"
Rhod-6-81, dated June 15, 1981.

3) Letter, D. W. Rhodes to W. B. Palmer, "Composition of
PEW Evaporator Feed, Condensate, and Bottoms," Rhod-17-83,
dated October 5, 1983.

Attached (Table I) is the data you requested concerning the current
average compositions and flow rates of the PEW Evaporator feed
streams. Table I lists the sources of the major evaporator feed
streams, as well as the chemical and radiochemical composition for
each stream. It should be understood by users of the attached table,
that the data base from which the flows and compositions was derived
was very limited. Most of the compositions came from the three
referenced letters, two of which are ~3 years old and may no longer
be representative of current operating conditions. Also, both
feed flow rates and compositions vary considerably throughout the
year. Some processes, such as the WC-114 evaporator, may be idle
for several months of the year and generate no waste at all during
that time; however, when operating, the waste flow rate sent to
the evaporator is much larger than that listed in the table. The
compositions of any given waste stream can also vary considerably.
The table lists the CPP-601 PEW/CFD tanks as a single source. Yet
samples from the four tanks taken during a single week of operation
(Ref 1) indicate the radionuclide compositions of individual tanks
can vary by two orders of magnitude. Since many of the waste streams
are not routinely sampled, it is difficult to tell if the isolated
samples from which the compositions in Table I were derived are
representative of average conditions. Due to the variability of
feed stream flows and compositions it is apparent that the average
flow shown in Table I will seldom, if ever exist, yet on a yearly
basis the table contains average numbers.

February 1, 1984
MCS-1-84
Page 2

The attached appendix contains a description of how each feed
stream's average composition was determined. Also included is a
simple check on the accuracy of the figures given in Table I.
This check simply applied average evaporator volume reduction and
decontamination factors to the figures given in Table I to calculate
a yearly nuclide release rate and evaporator bottoms generation rate.
These calculated numbers were then compared with actual values from
1982 and 1983 and found to agree quite closely (within a factor
of 3 or less).

rls

Attachment

Table I. Compositions and Volumes of PEW Evaporator Feed Streams

Source Stream	Volume (gal/day)	Chemicals (mg/L)				Radionuclides (μCi/ml)				
		Cl^-	F^-	Hg	NO_3^-	^3H	^{90}Sr	^{106}Ru	^{129}I	^{137}Cs
PEW/CFD Tanks (CPP-601)	1850	49	16	0.7	15100	0.001	0.1	0.1	0.16 E-5	0.1
CPP-603 Basin	350	195	0.9	0.002	2700	3.6 E-4	0.012	0	2.9 E-5	0.012
WC-119 (WCF evaporator condensate)	260	135	29	6.8	20600	0.26	0.1	0.01	17 E-5	0.016
Equip Decon (NWCF VES-123)	180	97	300	0.004	5600	1.5 E-4	5.7 E-4	0.0023	0.31 E-5	5.7 E-4
West Side Service Waste (WL-104)	90	83	1420	0.7	4600	0.003	0.019	0.019	0.55 E-5	0.019
NWCF (VES-122)	180	95	40	36	28000	0.12	0.1	0.5	2 E-5	0.1
Other INEL Sites (PBF,TRA,TAN)	260	20	1	0	5600	0.001	0.02	0	0	0.02
*APS Condenser	50	7	12	27	21000	0.35	0.05	0.38	1.3 E-5	0.39
Tank Farm Sumps	90	No Analyses Available								
**Other Misc.	1140	No Analyses Available								
***Totals	4450	75	71	3.4	13400	0.034	0.075	0.093	1.9 E-5	0.086

Annotation near ^{129}I column: 1.6×10^{-3} mg/L ; 1.6×10^{-2}

*Assumes a low (~2 gph) condensate rate even with condenser off.

**Misc. includes facility decons, RGP, various 604/605 CFD, recycled PEW condensate, diverted wastes, etc.

***Compositions are a volume weighted average.

Appendix

I. Derivation of chemical and radionuclide content of PEW Evaporator feed streams given in Table 1.

A) Feed stream flow rates are based on estimated yearly total flows averaged on a daily basis. Some of these flows may not even exist for months at a time but will have a much higher flow rate than that shown when operating. In general, the flow rates given agree (with 20%) of rates given in a letter Hamm-20-83/NPW-5-83 to G.L. Borsheim

B) Feed Stream Compositions

1) PEW/CFD Tanks (CPP-601) - Three samples were available (Ref. 1-3); 1981 during fuel processing and 1981, 1983 when fuel was not being processed. Assuming fuel processing to be in operation 75% of the time, the average PEW/CFD tank composition was calculated as follows:

$$[\ ]_{Ave} = [\ ]_{up} \times 0.75 + \frac{[\ ]_{down\,'81} + [\ ]_{down\,'83}}{2} \times 0.25$$

2) CPP-603 Basin - Two samples (1981, 1983) were available, but they differed so much (over one order of magnitude) that only the 1983 results were used. Even in the 1983 sample the ^{90}Sr and ^{137}Cs differed two orders of magnitude so the ^{90}Sr (higher of the two) result was used for both nuclides.

3) WG-119 (WG-114 evaporator condensate) - Two samples were available (1981 and 1983), but the analyses differed so much (two orders of magnitude) only the 1983 results were used.

4) Equipment Decon (VES-123 at NWCF) - Only one sample available (1983) so its results were used. However ^{90}Sr and ^{137}Cs differ so much the ^{137}Cs result was used on both

2

5) West Side Service Waste (WL-104) - Again two samples available (1981 and 1983) but used only 1983 results due to extreme differences. ^{137}Cs result was used for ^{90}Sr result (see reason on #4).

6) NWCF (VES-122) - No complete analyses are available on VES-122 usually only F, Cl, and gross β are requested. However, most of the solution in VES-122 is condensate from the NWCF off-gas intercoolers and should therefore be comparable in composition to the APS condenser condensate formed during past (WCF) operation. Therefore the 1981 APS condenser (OGF-104) results were used for Hg, NO_3, H-3, Ru-106, and I-129. I arbitrarily gave the same values as the CPP-601 PEW/CED tanks to both Sr-90 and Cs-137 (These values are higher than the APS condensate value, but VES-122 gross β indicate the solution to be "hot" and some decon solutions go to VES-122). The Cl and F were average results of actual NCC-122 samples.

7) Other INEL Sites: looked at several analyses for past shipments from TRA, PBF, and TAN; used the average results for Cl, F, and Cs-137; Ru-106 and I-129 were not detected so listed them as "0"; Gave Sr-90 the same value as that of Cs-137 (Sr usually not requested in analysis); Hg also not requested on analysis and probably (?) not present. Since most of the solutions are decon, arbitrarily gave H-3 the same result as CPP-601 PEW/CED tanks and NO_3 the same as NCD-123.

8) APS Condenser (OGF-104) - Although cooling water is usually off, there is likely a slow (2 gph) condensate formation. Assuming both NWCF and fuel dissolvers are on-line 75%, I took the higher of the two 1981 samples' values (one with WCF up, one with fuel processing up) for each component

3

9) Tank Farm Sumps — No sample analysis available. Normally the solution should be mostly water (due to seepage etc into vaults) but due to past inadvertent/incorrect transfers, some vaults and sumps may contain a considerable amount of activity.

10) Misc. Due to the variety of liquid sources no attempt was made to estimate "average" compositions

11) TOTALS: The compositions shown are weighted averages calculated as follows;

$$[A]_{ave} = \frac{1 \times [A]_1 + 2[A]_2 + 3[A]_3 + \ldots + [A]_n}{1 + 2 + 3 + \ldots + n}$$

where $[A]$ is concentration of element (or isotope) A and $1, 2, 3$ etc refer to flow rates (gpd) of streams $1, 2, 3$ etc

4

Accuracy Check of PSW compositions/flows

1) Use Average volume Reduction and decontamination factors to calculate releases and compare to actual releases

Assume Volume Reduction Factor (VRF) ~ 65 = $\frac{\text{Volume overheads}}{\text{Volume Bottoms}}$

(DF) Decontamination Factors = $\frac{\text{Concentration in Bottoms}}{\text{Concentration in Overheads}}$ (variable)

2) Sample Calc:

^3H – Assume DF=1 ; 4450 gph with 0.034 μCi/ml

^3H to well = $\left(4450 \frac{gal}{day}\right)\left(365 \frac{days}{yr}\right)\left(0.034 \frac{\mu Ci}{ml}\right)\left(3785 \frac{ml}{gal}\right)\left(\frac{1 Ci}{10^6 \mu Ci}\right)$ = 209 $\frac{Ci}{yr}$

209 Ci calculated vs. 295 in 1982
(478 projected in 1983)

^{90}Sr assume DF=1×10⁶ ; 4450 gph with 0.075 μCi/ml

^{90}Sr to well = $\left(4450 \frac{gal}{day}\right)\left(365 \frac{1 day}{yr}\right)\left(0.075 \frac{\mu Ci}{ml}\right)\left(3785 \frac{ml}{gal}\right)\underset{VRF}{\left(\frac{65}{1}\right)}\underset{DF}{\left(\frac{1}{1\times10^6}\right)}\left(\frac{1 Ci}{10^6 \mu Ci}\right)$ = 0.030

0.030 calculated vs 0.158 in 1982
0.038 in 1983 projected

Similarly The following table was developed:

Nuclide	Ci to Service Waste* 1981	1982	1983**	Calculated Ci	Calculation Assumptions***
^3H	359	295	478	209	DF=1
^{90}Sr	0.207	0.158	0.0384	0.030	DF=10⁶ VRF=65
^{106}Ru	1.05	3.53	0.397	0.400	DF=10⁵ VRF=65
^{129}I	0.044	0.015	0.048	0.12	DF=1
^{137}Cs	0.419	0.390	0.0614	0.034	DF=10⁶ VRF=6

* Table is based on corrected data received from QASES on 11-3-83
** Extrapolated based on 9 months of 1983 data
*** Typical DF's and VRF's were used

3) Bottoms generation calculation: (VRF=65)

$\sim$ 1) Bottoms = $\left(4450 \frac{gal}{day}\right)\left(30 \frac{day}{}\right)\left(\frac{1}{}\right)$ = 2054 ml $\Rightarrow$ 1982 BOTTOMS = 2112 gal
1983 BOTTOMS = 2085

B-23